DOE/NE-ID-11226
Revision 0 Draft

Draft Section 3116 Determination

Idaho Nuclear Technology and Engineering Center Tank Farm Facility

September 7, 2005

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ACRONYMS

ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
ASTM	ASTM International (formerly American Society for Testing and Materials)
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CWI	CH2M-WG Idaho, LLC
DCF	dose conversion factor
DEQ	State of Idaho Department of Environmental Quality
DOE	U.S. Department of Energy
DOE Idaho	U.S. Department of Energy Idaho Operations Office
DQA	data quality assessment
DQO	data quality objective
DUST-MS	Disposal Unit Source Term-Multiple Species
EPA	U.S. Environmental Protection Agency
ESRP	Eastern Snake River Plain
HLW	high-level waste
HWMA	Hazardous Waste Management Act
ICRP	International Commission on Radiological Protection
IDAPA	Idaho Administrative Procedures Act
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LLW	low-level waste
MCP	management control procedure

M&O	management and operating
NDAA	National Defense Authorization Act
NRC	Nuclear Regulatory Commission
PA	performance assessment
PEWE	Process Equipment Waste Evaporator
QA	quality assurance
RCRA	Resource Conservation and Recovery Act
RPP	radiation protection program
SAP	sampling and analysis plan
SBW	sodium-bearing waste
SNF	spent nuclear fuel
SSC	structure, system, and component
TEDE	total effective dose equivalent
TFA	Tanks Focus Area
TFF	Tank Farm Facility
USC	United States Code
WAC	waste acceptance criteria

Draft Section 3116 Determination
Idaho Nuclear Technology and Engineering Center
Tank Farm Facility

1. INTRODUCTION AND PURPOSE

Key Points

- This draft 3116 Determination addresses the proposed disposal of grouted waste residuals in the TFF tank system and disposal of the tanks, vaults, and associated piping and ancillary equipment at INTEC. The TFF tank system comprises the 11 300,000-gal tanks, four 30,000-gal tanks, and the vaults, piping, structures, and ancillary equipment associated with these tanks.
- The scope of this draft 3116 Determination does not include any other facilities or systems at INTEC or the INL Site such as the evaporators, the New Waste Calcining Facility, the calcine storage bin sets, the 900,000 gal of stored sodium-bearing waste, contaminated soils, or the treated sodium-bearing waste for off-Site disposal.
- The planned TFF closure date of 2012 is assumed throughout this draft 3116 Determination.
- The TFF tank system closure process includes cleaning activities and stabilization activities. After the sodium-bearing waste is removed, the tanks, vaults, piping, structures, and ancillary equipment will be cleaned. Then the system will be stabilized by filling the components with grout. This distinction between cleaning activities and stabilization activities (grouting) applies throughout this document. The term "final closure" is used to describe the completion of both cleaning and stabilization activities.
- This draft 3116 Determination addresses only radioactive constituents of the TFF residual waste; hazardous constituents of this waste are not addressed.

This draft 3116 Determination is being issued pursuant to Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) (Public Law 108-375, 2004). This document concerns the disposal of grouted residual waste in the tank systems at the Idaho Nuclear Technology and Engineering Center (INTEC) Tank Farm Facility (TFF) that are being closed in accordance with U.S. Department of Energy (DOE) requirements for managing a radioactive waste storage facility (DOE O 435.1, 2001) and Hazardous Waste Management Act (HWMA) (State of Idaho 1983)/Resource Conservation and Recovery Act (RCRA) (42 United States Code [USC] 6901 et seq., 1976) requirements for an interim status tank system. The DOE is closing the TFF tanks in response to a January 1990 Notice of Noncompliance and subsequent Consent Order (State of Idaho 1992). The TFF consists of 11 300,000-gal belowgrade stainless steel tanks in unlined concrete vaults, four 30,000-gal belowgrade stainless steel tanks, and associated ancillary equipment and piping. Historically, the TFF tanks were used to store various INTEC wastes, including those from spent nuclear fuel (SNF) reprocessing (first-, second-, and third-cycle reprocessing wastes), decontamination waste, laboratory waste, and contaminated liquids from other INTEC operations. In general, because of significantly higher radioactivity levels, first-cycle reprocessing wastes were segregated from the other types of liquid waste. These other tank wastes, referred to as sodium-bearing waste (SBW) because of their high sodium levels, were made up of wastes other than first-cycle reprocessing wastes, were generally much lower in radioactivity, and had a significantly different chemical composition than first-cycle reprocessing wastes.

Spent nuclear fuel reprocessing was curtailed in 1992, and the first-cycle extraction process wastes stored in the TFF were removed and solidified (by calcination) by February 1998 (DOE 2002).[a] Since that time, the tanks used for storing the first-cycle wastes have been reused to store SBW. Because of the design of the tank's originally installed waste removal system, a tank emptied using this system still contains several thousand gallons of waste on the bottom. Thus, the reuse of the TFF tanks that were used to store first-cycle extraction process wastes to store SBW, along with the various historical transfers of wastes among TFF tanks to manage TFF volumes, has resulted in the current volume of SBW containing a small percentage (about 1% by volume) of first-cycle extraction process wastes (Loos 2004).

The TFF is being closed in phases to support continued INTEC operations. The closure process comprises tank system cleaning and stabilization activities. Over the last several years, TFF operations have included consolidating the remaining SBW tank wastes into the minimum number of tanks necessary and commencing cleaning activities in the emptied tanks. Tank cleaning began in late 2002, and the project has progressed on schedule. Final closure is planned for 2012. As of July 2005, seven of the 300,000-gal tanks, the four 30,000-gal tanks, and associated ancillary equipment have been cleaned. Approximately 900,000-gal of SBW remain stored at the TFF awaiting future treatment. This SBW is stored in three 300,000-gal tanks; one 300,000-gal tank is maintained as a spare.[b] After completing tank and ancillary equipment cleaning operations, a small amount of residual radioactive waste that cannot be removed remains. As each tank is cleaned, these residuals are sampled and analyzed to confirm that radionuclide and hazardous constituent concentrations meet performance objectives to ensure protection of the public and the environment. After cleaning activities are completed for all of the tanks and ancillary equipment in the TFF, the DOE plans to stabilize the TFF by filling the tank system with grout for final closure of the TFF. Throughout this document, grouting (stabilization) activities are distinct from cleaning activities and not considered part of the cleaning process. The term "final closure" is used to describe the completion of both cleaning and stabilization activities.

Section 3116 of the NDAA specifies that the term "high-level radioactive waste" does not include radioactive waste that results from reprocessing SNF if the Secretary of Energy determines, in consultation with the Nuclear Regulatory Commission (NRC), that the waste meets certain criteria, which specify that the waste must not require disposal in a deep geologic repository; must have had highly radioactive radionuclides removed to the maximum extent practical; must meet certain concentration limits and performance objectives for low-level waste (LLW); and must be disposed of pursuant to a State-approved closure plan or State-issued permit. In this document, the Secretary of Energy proposes to determine that the TFF waste residuals, and the tanks, vaults, and associated piping, structures, and equipment after final closure activities are completed will meet all of these criteria. Accordingly, these wastes may be determined not to be high-level waste (HLW) and may be grouted and disposed of in place as LLW. For the purposes of this draft 3116 Determination, the TFF tank system comprises the 11 300,000-gal tanks, four 30,000-gal tanks, and the vaults, piping, structures, and ancillary equipment associated with these tanks. The scope of this draft 3116 Determination does not include any other facilities or systems at INTEC or the Idaho National Laboratory (INL) Site such as the evaporators or the New Waste Calcining Facility.

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a. This calcined waste is planned to be disposed of in the geologic repository at Yucca Mountain, Nevada, after the repository is licensed and operational. The calcined waste is not addressed in this draft 3116 Determination because, among other things, Section 3116(c), in conjunction with 3116(d), provides that Section 3116 does not apply to waste transported from the State of Idaho.

b. The SBW, which will be or has been removed from the tanks, and treated for off-Site disposal is not part of this draft 3116 Determination because, among other things, Section 3116(c), in conjunction with 3116(d), provides that Section 3116 does not apply to waste transported from the State of Idaho.

To demonstrate that the TFF waste residuals and associated ancillary equipment at final closure will meet the Section 3116 criteria, historical waste management information, performance assessment (PA) results, and sampling and analysis results from the recent tank cleaning activities were reviewed and analyzed. In addition, the residual waste inventory at closure was updated to reflect the results of TFF cleaning activities. Prior to cleaning and receiving sampling and analysis results of the first tank cleaning, planning documents such as the first HWMA/RCRA closure plan (DOE-ID 2003a) and the TFF PA (DOE-ID 2003b) were prepared. These planning documents determined the baseline inventory and consequently established the goals for the cleaning activities to meet or exceed. The total post-decontamination inventory for each of the cleaned TFF tanks is significantly less than the total conservative post-decontamination inventory of a single tank estimated in the PA (24,103 Ci), indicating that the tank cleaning operations performed since this baseline inventory was established performed better than planned, as presented in Appendix A. The greatest inventory for a cleaned tank is Tank WM-182, which contains 2,393 Ci.

The purpose of this draft 3116 Determination is to demonstrate and document that the grouted TFF residuals and the tanks, vaults, and associated piping and ancillary equipment meet the 3116 criteria, and therefore, may be determined not to be HLW and may be disposed of as LLW at INTEC in accordance with Section 3116 of the NDAA (Public Law 108-375, 2004). This document addresses only radioactive constituents in the INTEC TFF after final closure. Characterization of the hazardous constituents of the waste residuals in accordance with RCRA is not addressed in this document. The State of Idaho, which has primacy for RCRA compliance but does not have regulatory oversight of the radioactive materials contained in the TFF tanks and ancillary equipment, approves the HWMA/RCRA closure plans.[c]

This draft 3116 Determination will be finalized after DOE has completed consultation with the NRC. Although not required by Section 3116, DOE is also issuing this draft 3116 Determination for public review and comment.

c. Decisions regarding past releases of contaminants and the impacts of contaminated soils associated with the TFF will be addressed under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (42 USC 9601 et seq., 1980) process as specified in the Federal Facility Agreement and Consent Order among DOE, the State of Idaho, and the U.S. Environmental Protection Agency (EPA) (State of Idaho 1991).

2. BACKGROUND

Key Points

- The residual waste inventory at closure is an estimate of the amount of radioactivity remaining in the residual in the TFF at closure in 2012, the planned closure date. All radionuclide concentrations are decayed to 2012.
- The residual waste inventory for the 11 300,000-gal tanks is based on analytical data from samples obtained after cleaning operations from the seven tanks that have been cleaned.
  - The inventory for the 300,000-gal tanks assumes (for the reasons explained in Subsection 2.4.2) that the same degree of radionuclide removal will be achieved in the four tanks remaining to be cleaned as that achieved in the seven tanks that have been cleaned.
  - Residual solids are estimated from a sample of solids collected from Tank WM-183. Concentrations for radionuclides not detected in this sample are estimated using the ORIGEN2 numerical model and ^{137}Cs concentrations in Tank WM-188 samples collected prior to cleaning.
  - The inventory is not reduced by the amount of additional residual waste that may be removed from the cleaned tanks during grouting operations.
- A 5-mil (0.005-in.) thick film of residual solids is assumed for the 30,000-gal tank inventory, for the reasons described in Subsection 2.4.3. Radionuclide concentrations are assumed (for the reasons described in this draft 3116 Determination) to be that found in the Tank WM-183 solid samples.
- The residual waste inventory for the sandpads underlying Tanks WM-185 and WM-187 is a conservative inventory representing the sandpad under these tanks in the PA inventory. The inventory is based on the concentrations of radionuclides in liquid waste that was siphoned into the Tank WM-185 vault in 1962.
- The piping residual waste inventory is based on analytical results from sampling pipe sections removed from the Tank WM-182 process waste lines after cleaning. The inventory for valves is included with the piping inventory. No inventory is calculated for valve boxes, pipe encasements, and off-gas piping because these components did not normally contain process solutions.
- This draft 3116 Determination addresses disposal of the TFF residuals and the TFF tank system at closure, when the entire TFF has been cleaned and grouted as described in this document. The closure is expected to be completed by 2012 and decay is calculated by that date. But, as described in more detail in this draft 3116 Determination, differences in decay between the present time and 2012 are not significant (and do not affect compliance with the performance objectives of 10 CFR 61, Subpart C as considered in this draft 3116 Determination).
- Modeling is used to estimate concentrations for radionuclides for which analytical data are not available or for which radionuclides were not detected in samples. Modeling and calculations are based on the planned 2012 closure date.
- The sandpads are structural components of the TFF and are located inside the vault and under the 300,000-gal tanks. The vaults of Tanks WM-182 through WM-190 contain sandpads.

The INTEC TFF is located on the INL Site. The INL is an approximately 2,305-km^2 (890-mi^2) reservation owned by the United States government and located in southeastern Idaho (see Figure 1). Established over 50 years ago as the National Reactor Testing Station, INL's initial mission was to develop civilian and defense nuclear reactor technologies. Over the years, the INL mission evolved beyond the original reactor technology focus. The INL is now involved in various defense, energy supply, industrial technology, and environmental programs. In recognition of this evolution to a multi-program installation, the site was designated the Idaho National Engineering Laboratory (INEL) in 1974. In January 1997, the name was changed to the Idaho National Engineering and Environmental Laboratory (INEEL) to reflect greater emphasis on the laboratory's environmental missions. In February 2005, the INEEL became two separate organizations: INL, which performs its continuing research mission, and Idaho Cleanup Project, which carries out the site's cleanup responsibilities.

In 1953, the Idaho Chemical Processing Plant, now INTEC, was chartered to recover fissile uranium by reprocessing SNF. In 1992, the DOE officially discontinued reprocessing SNF at INTEC. This decision changed INTEC's mission to managing, storing, and treating reprocessing wastes generated from past and current operations and activities. The INTEC facility is located approximately 29 km (18 mi) from the closest eastern boundary, approximately 23 km (14 mi) from the closest western boundary, approximately 16 km (10 mi) from the closest southern boundary, and approximately 29 km (18 mi) from the closest northern boundary. The TFF, located within the northern portion of INTEC (see Figures 2 and 3), comprises 11 300,000-gal belowgrade stainless steel tanks[d] in unlined concrete vaults of various construction, four inactive 30,000-gal stainless steel tanks, interconnecting waste transfer lines, and associated support instrumentation and valves. Structures located above ground level in the TFF include the TFF Control House, the Computer Interface Building, and the tank and vault sump riser covers. A perimeter fence encloses the TFF. Subsection 2.1 provides a detailed description of the TFF. Subsection 2.2 presents a discussion of the wastes stored in the TFF and waste management practices.

Recent tank cleaning operations have resulted in the removal of the remaining SBW and tank heels from seven 300,000-gal tanks and four 30,000-gal tanks to the maximum extent practical for the cleaned tanks. Four 300,000-gal tanks remain to be cleaned, and these four tanks are anticipated to be cleaned as efficiently as the other 300,000-gal tanks that have been cleaned. Sampling and analysis of tank and ancillary equipment residuals indicate that the residual waste inventory at closure is less than that established in the 2003 PA (DOE-ID 2003b). The residual waste inventories at closure in a stabilized form are considered in demonstrating that the TFF tank system residual waste at final closure will meet Section 3116 criteria. Subsection 2.3 describes the TFF closure activities and status, including the tank cleaning equipment. (A summary of the tank cleaning technology evaluation is presented in Appendix B.) The residual waste inventory at closure and the characterization of the TFF waste residuals after tank cleaning are presented in Subsection 2.4.

2.1 Tank Farm Facility Description

Placed into service between 1953 and 1966, the 11 300,000-gal tanks (WM-180 through WM-190) are approximately 15.2 m (50 ft) in diameter and 6.4–7.0 m (21–23 ft) in height. Nine of the 11 300,000-gal tanks are constructed of Type 304L stainless steel; two tanks (WM-180 and WM-181) use Type 347 stainless steel. Each tank has four or five 30-cm (12-in.) diameter risers to provide access to the tank. Tanks WM-184 through WM-190 also have one or two 46-cm (18-in.) diameter risers. Most risers have installed equipment and instrumentation, including radio frequency probes for level measurement,

d. Tanks WM-180 and WM-181 are actually 318,000-gal tanks; however, for ease of reference, all 11 tanks will be referred to as 300,000-gal tanks.

corrosion coupons,[e] and steam jets and airlifts for waste transfer operations. Two steam jets are located inside each tank, except for Tanks WM-189 and WM-190, which have one steam jet and one airlift pump. High-pressure steam is forced through a steam jet to create a suction pumping action to move liquid from the tank to Tank WM-187 until final treatment and disposition are identified.

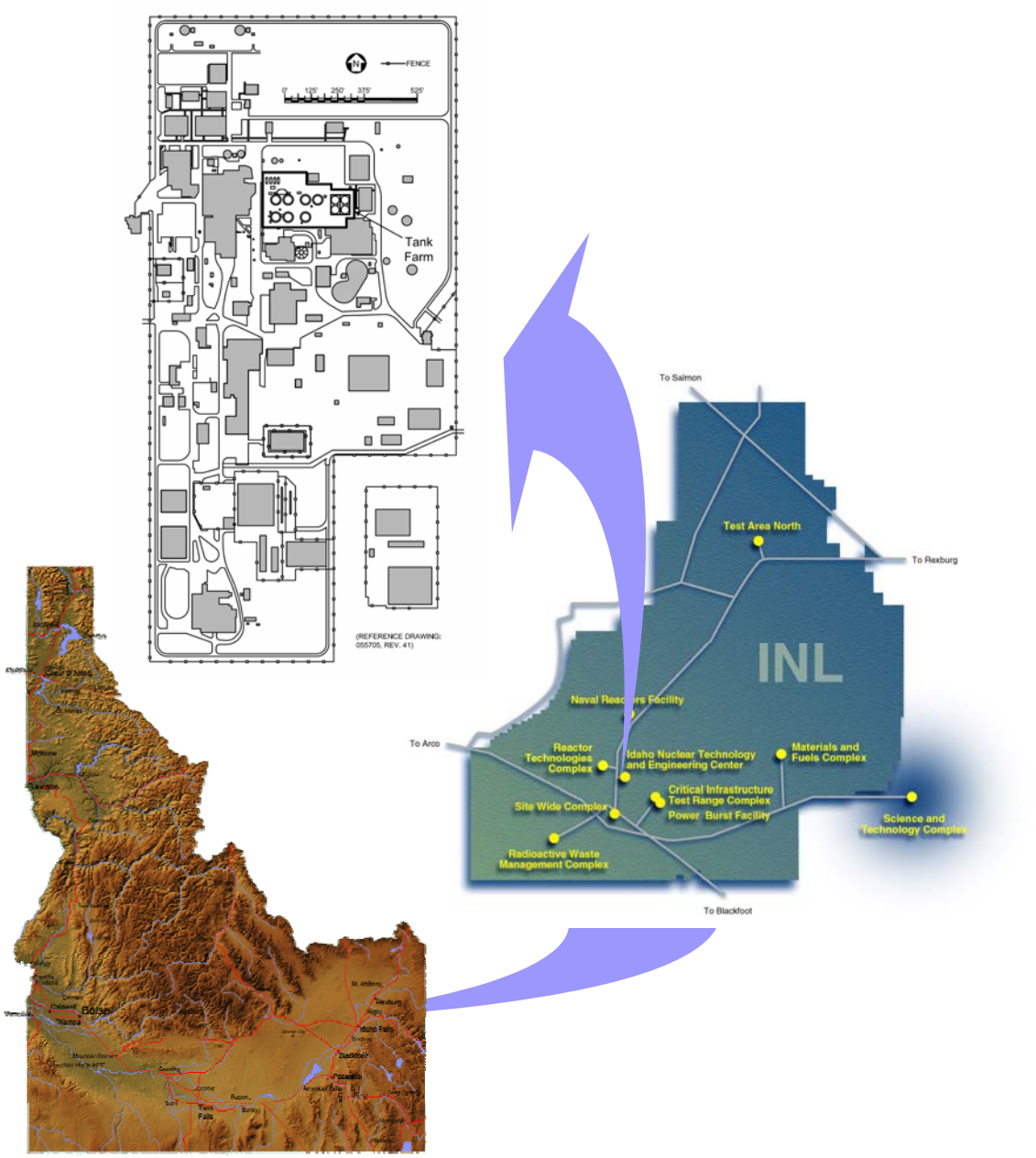

Figure 1. Idaho National Laboratory site map.

e. A corrosion coupon is a piece of material, in this case metal, of known weight used to monitor corrosion in a specific environment. The coupon weight before and after exposure is measured to determine loss from corrosion. The condition of the coupon may also be observed.

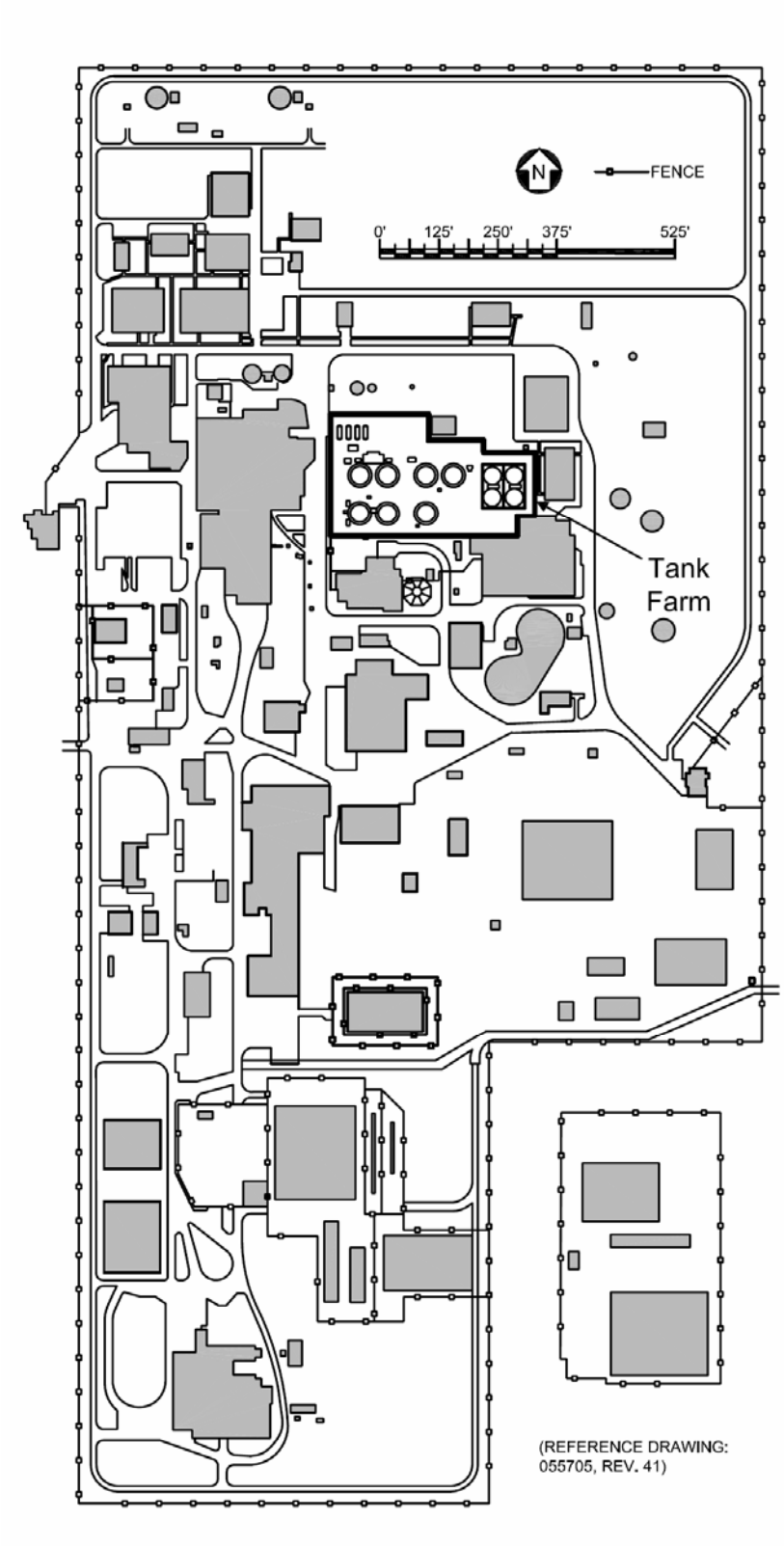

Figure 2. Tank Farm Facility location at INTEC.

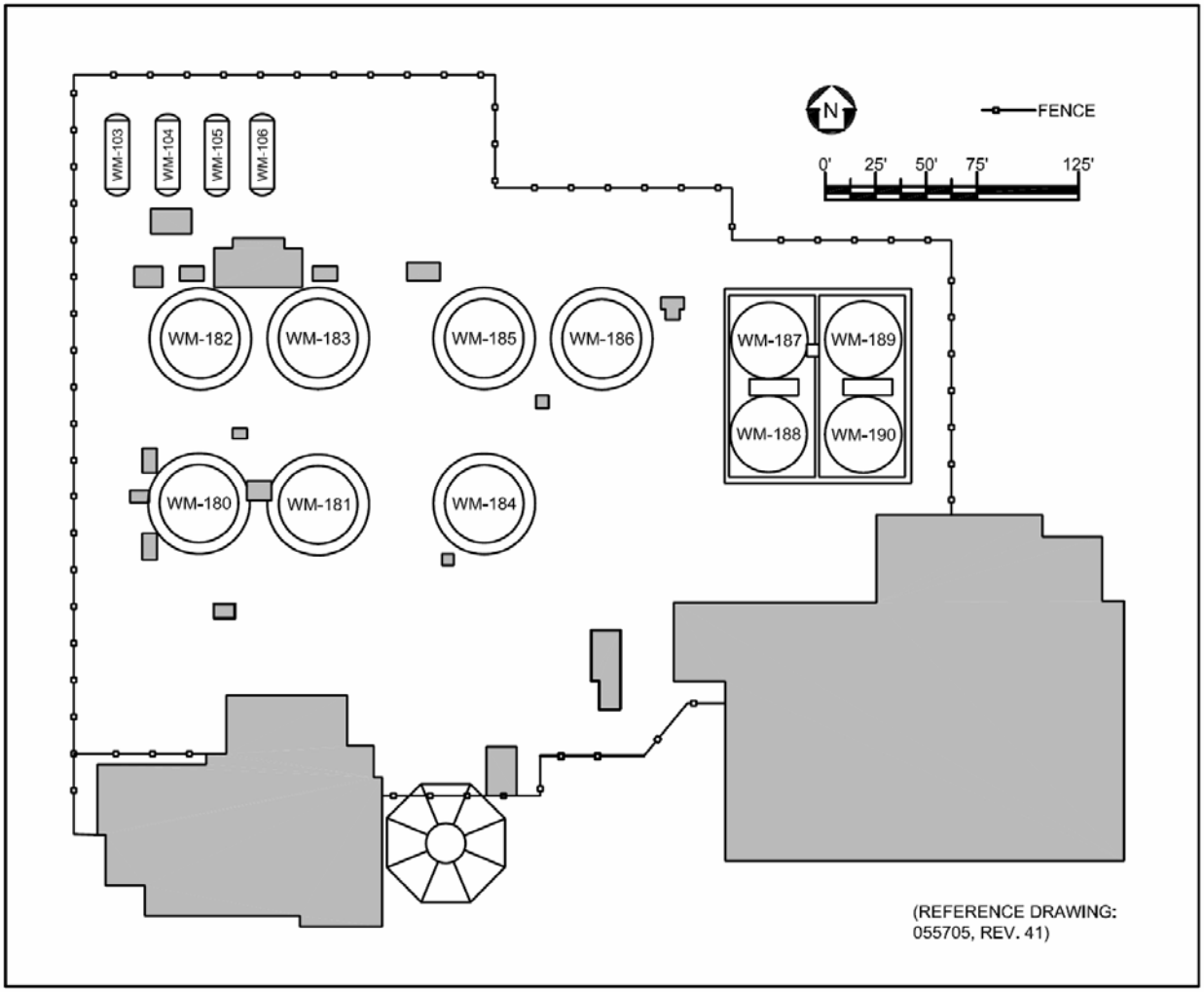

Figure 3. Plan view of the Tank Farm Facility.

A single steam-jet pump can transfer waste out of a tank at approximately 50 gpm. The original design of the suction pumps restricts the ability of the pumps to clear the tanks of liquid completely. Using original steam jet placement and prior to any tank cleaning activities, an 8–30-cm (3–12-in.) deep residual containing both liquids and solids remains on the floor of the tanks. Therefore, steam-jet pumps are lowered to the bottom of the tank for final tank cleaning. Figure 4 presents the dimensions and structure of a typical 300,000-gal tank. The 300,000-gal tanks are housed in concrete vaults approximately 13.7 m (45 ft) below grade level. The vaults have one of three different designs: (1) octagonal pillar-and-panel vaults (five tanks), (2) cast-in-place square vaults (four tanks), and (3) cast-in-place octagonal vaults (two tanks). Figure 5 depicts the different vault designs. Cooling coils on the floor and walls of eight of the tanks (WM-180, WM-182, WM-183, WM-185, and WM-187 through WM-190) provide heat transfer capabilities. Figure 6 is a photograph showing the cooling coils in Tank WM-185 during construction. The 30,000-gal tanks and 15-cm (6-in.) thick concrete vault roofs are covered with approximately 3 m (10 ft) of soil to provide radiological protection to workers and the public.

When the TFF was constructed, the 300,000-gal tanks were designed with flat bottoms. The concrete floors of the tank vaults were designed with sloped floors to promote drainage of any liquids toward the perimeter of the vaults for efficient removal. In order for the tank to rest on the sloped floor without causing unacceptable stresses in the tank, a leveling pad of sand was installed to support the tanks. The sandpad is 15 cm (6 in.) thick at the perimeter and 5 cm (2 in.) thick at the center. The exterior of the sandpad extends 15 cm (6 in.) beyond the outer edge of the tanks and is confined by a curb measuring 15 by 15 cm (6 by 6 in.). The volume of the sandpad is 23.39 m^3.

Constructed in 1954, the four inactive 30,000-gal stainless steel belowgrade storage tanks (WM-103 through WM-106) sit on reinforced concrete pads. The tanks are horizontal cylinders approximately 3.5 m (11.5 ft) in diameter and 11.6 m (38 ft) in length. The 30,000-gal tanks do not have vaults. All four tanks contain stainless steel cooling coils to provide heat transfer capabilities. Three 15-cm (6-in.) diameter risers and one 8-cm (3-in.) diameter riser that reach to grade level provide tank access. Tank steam jets are provided for liquid waste removal (nominal flow rate of 50 gpm). These tanks were removed from service in February 1983.

Figure 7 is an artist's representation of the 11 300,000-gal tanks (WM-180 through WM-190) and the four 30,000-gal tanks (WM-103 through WM-106) to be included in the INTEC TFF closure. Liquid waste transfers to, from, and among the tanks are managed through a system of lines, valves, and diversion boxes. Waste transfer lines are contained within one of the following types of secondary encasements: split tile, carbon steel, stainless steel-lined concrete troughs, double-walled stainless steel pipes, or buried directly in concrete. Because of upgrades over the years, the pipe sections encased with the carbon steel and split tile have either been replaced or abandoned in place, except for two small (about 1.5-m [5-ft]) sections that are no longer used. The majority of the upgraded piping is contained in stainless steel-lined concrete trenches with the remainder in double-walled stainless steel piping. Double-walled stainless steel piping was used primarily in areas where single pipe runs were upgraded or where access did not allow lined trench installation. Liquid waste is routed through waste transfer valves located in belowgrade, stainless steel-lined concrete boxes (referred to as valve boxes). The waste transfer valves are operated manually using reach rods. The valve boxes are designed to provide access to the valves for inspection and maintenance.

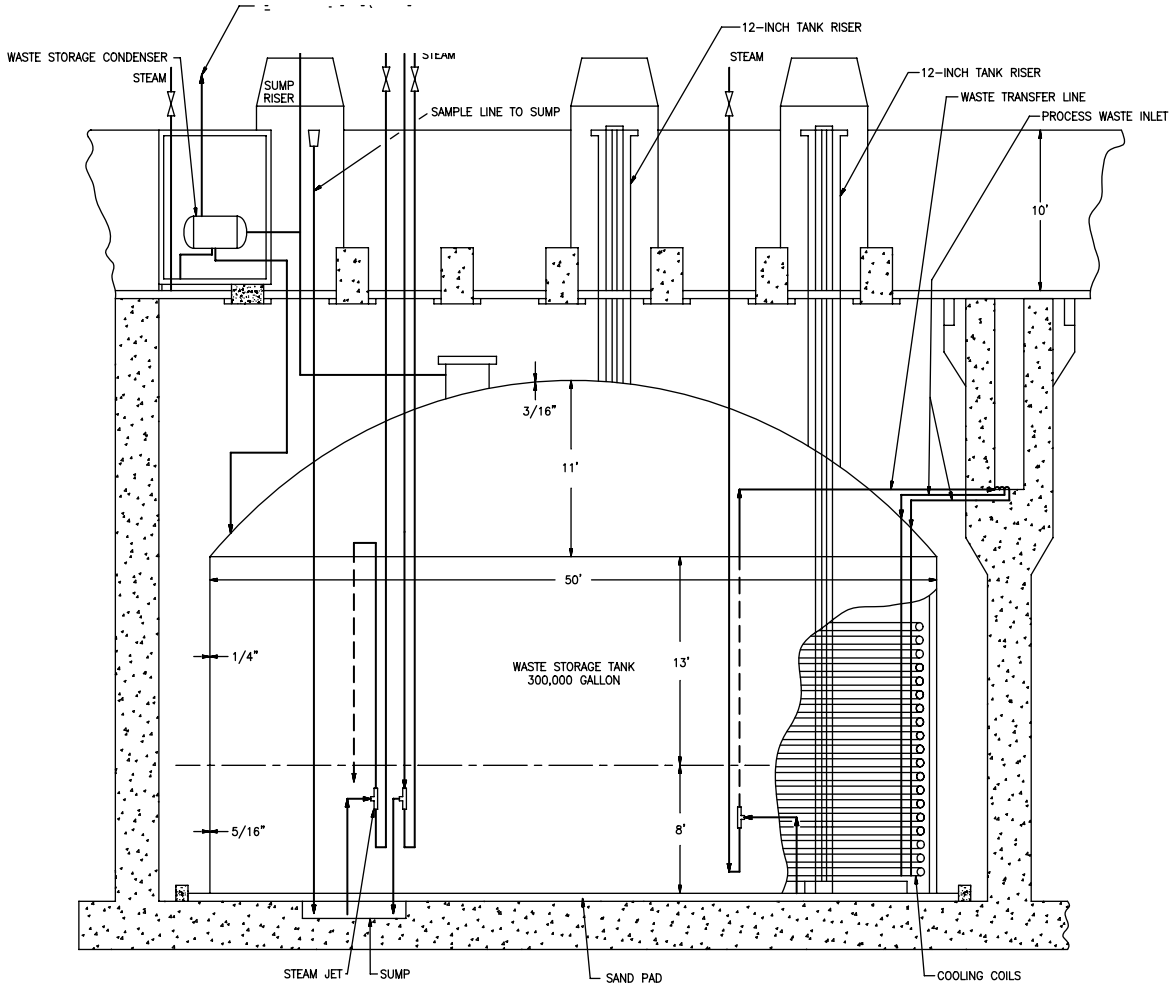

Figure 4. Cross-sectional view of a typical tank with cooling coils.[f]

f. This schematic is labeled as typical because it shows a configuration for the 300,000-gal tanks in the TFF. Not all features of each type of tank are shown but features such as the cooling coils, risers, and sandpad are shown in this schematic.

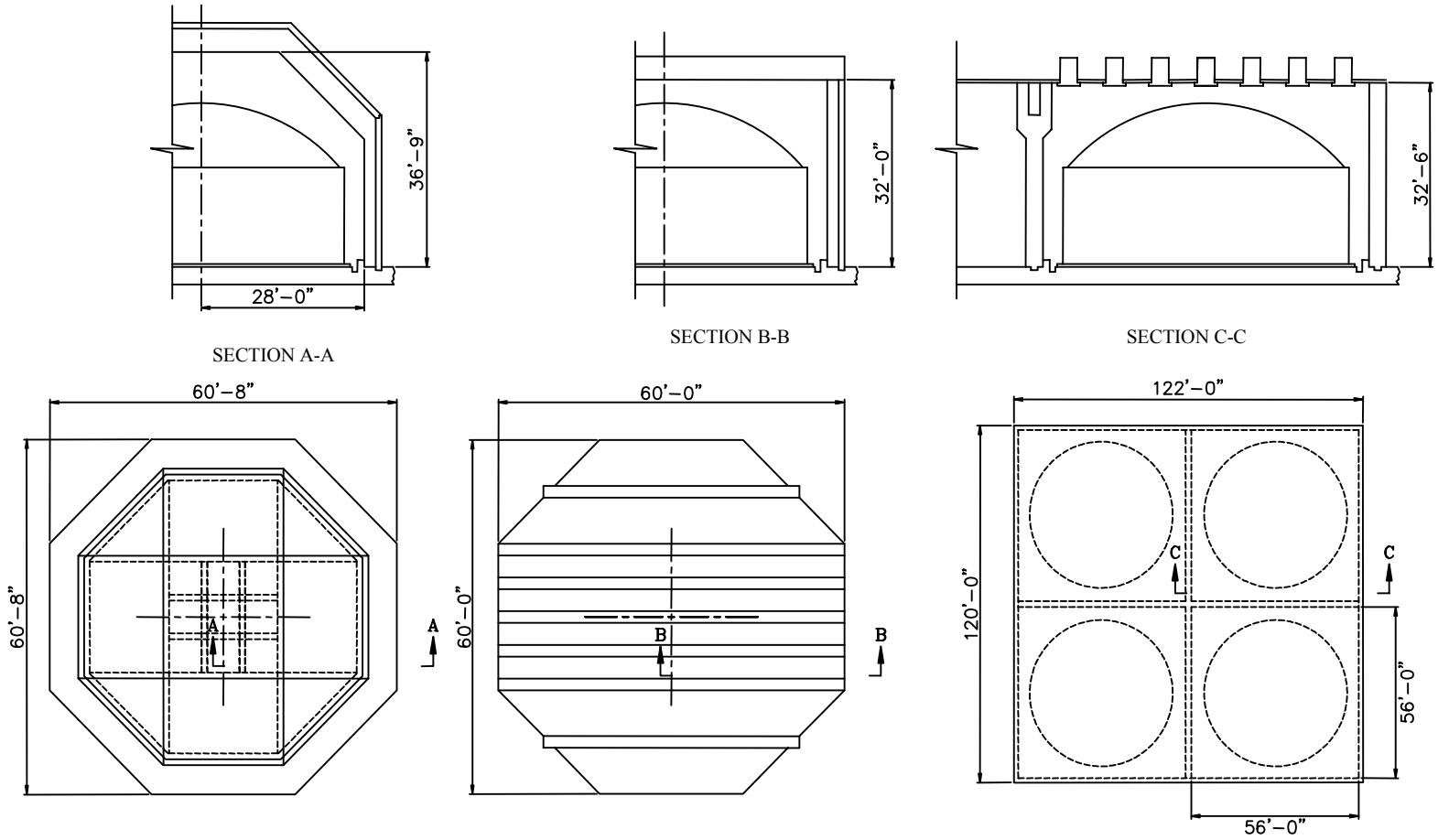

Figure 5. Vault designs used at the INTEC TFF.

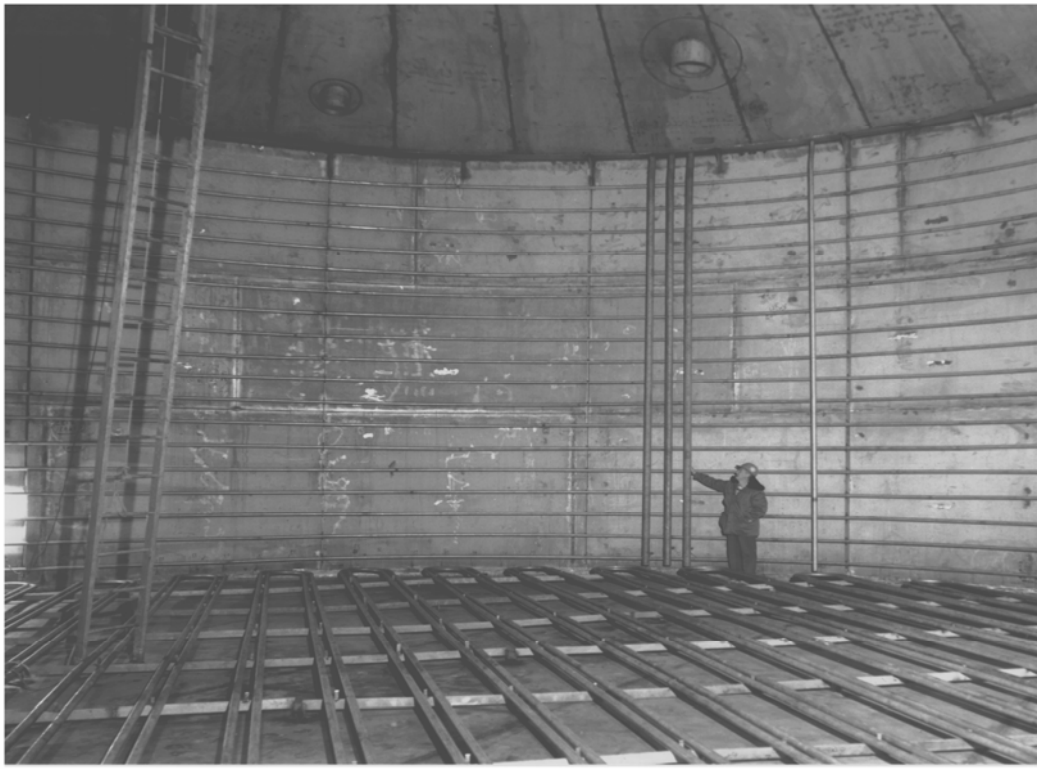

Figure 6. Photograph showing the internal structure and cooling coils of Tank WM-185 during construction.

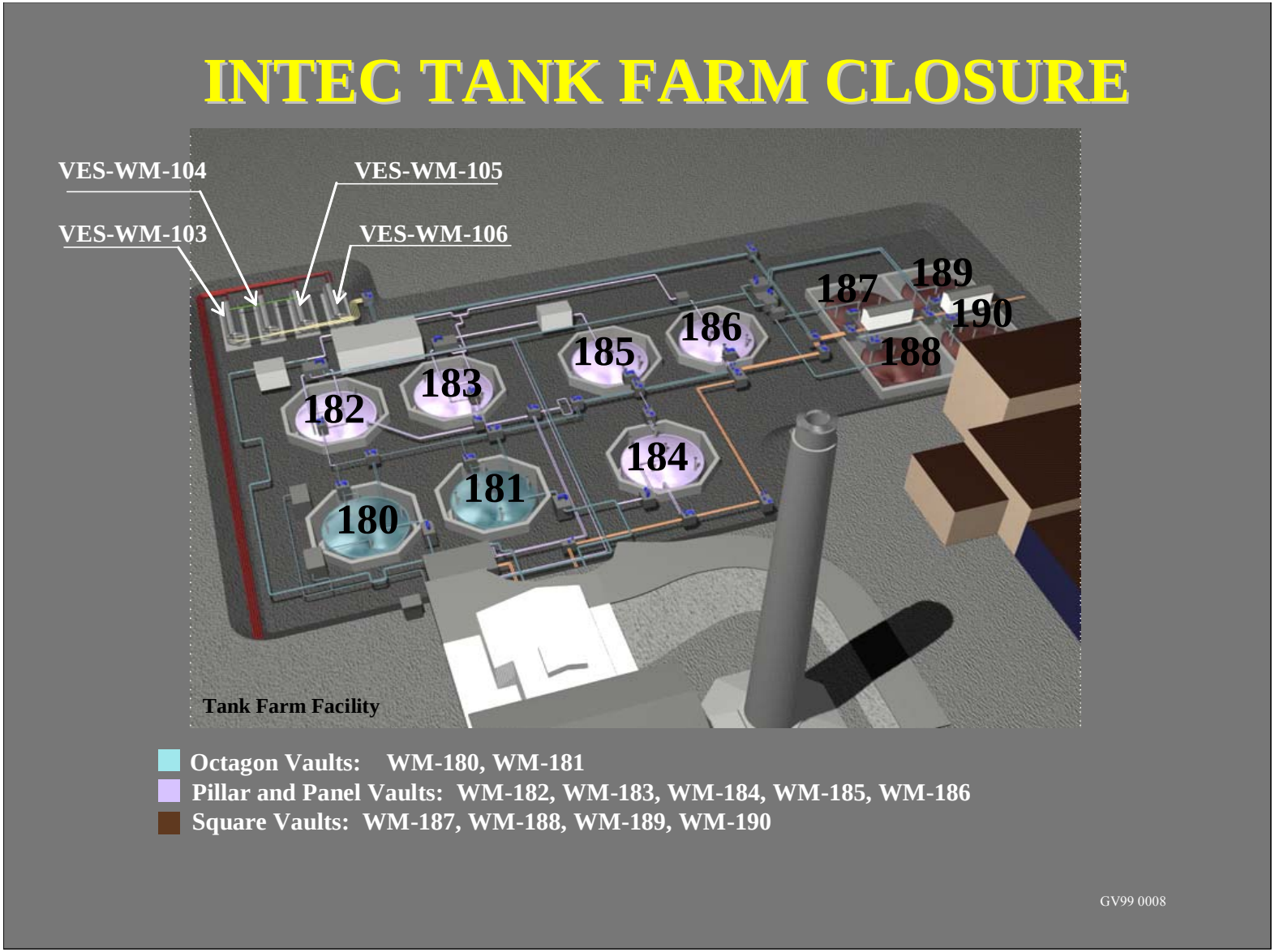

Figure 7. Artist's depiction of tanks to be included in the INTEC TFF closure.

2.2 Origin and Management of TFF Wastes

During reprocessing operations between 1953 and 1992, spent fuel was dissolved at INTEC using various processes, depending on the fuel type (see Figure 8).[g] Each dissolution process produced an acidic aqueous solution (any caustic dissolution processes were adjusted to be acidic before further treatment). The aqueous solution was processed through the first-cycle extraction system (Phase I in Figure 8) with an organic solvent (usually tributyl phosphate in kerosene). The extraction system used several contactors, including pulsed-plate columns and packed columns. The uranium was partitioned from the bulk of the fission products and placed in intercycle storage to await purification. The aqueous waste phase that contained greater than 99% of the radioactive material was the first-cycle extraction waste (Loos 2004), which was generally stored in the TFF in the belowgrade tanks equipped with cooling coils because of significant heat generation from the high radioactivity levels in the waste.

Typical of liquid-liquid extractions, the separated uranium contained some carryover radioactive material. After sufficient product accumulated in the intercycle storage, the uranium was processed through the second- and third-cycle extractions (Phase II in Figure 8), where the excess radioactive material was removed to produce a clean uranium product.[h]

The second and third fuel reprocessing steps (called the second and third cycles) generated the smallest volumes of waste of the five major waste sources. The second- and third-cycle processes were uranium purification steps and were very similar to each other. The second- and third-cycle processes operated together (in series) and were closely coupled. The second cycle purified the uranium product from the first-cycle process in a liquid extraction system. The second-cycle process produced a purified aqueous uranium product and a waste stream (second-cycle raffinate) containing radioactive contamination. The third cycle was an additional purification step that provided further purification of the second-cycle uranium product in a liquid extraction system, similar to that of the second cycle. The third-cycle process produced a further purified uranium product and a waste stream (third-cycle raffinate) containing radioactive contamination. Originally, the purified aqueous uranium product from the third-cycle process was shipped to Oak Ridge National Laboratory. Later, a plant modification provided a solidification process, the denitrator, to convert the aqueous uranium product into a solid granular form for shipment to Oak Ridge National Laboratory (Loos 2004).

The original SNF reprocessing system combined the second- and third-cycle wastes into a single waste stream for storage in the TFF, due to their similarity in chemical and radionuclide content. The combined second/third-cycle raffinate was stored separately from first-cycle waste in the TFF. The activity of the second/third-cycle waste was low enough that it did not require cooling. Consequently, second/third-cycle waste was originally stored in tanks without cooling capability (Loos 2004).

Unlike the first-cycle waste, the composition of the second/third-cycle waste varied little with the type of fuel being processed. Chemicals unique to various first-cycle wastes that came from the fuel dissolution process, such as Zr, F, Cd, and Hg, were separated from the uranium product and went with

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g. Four fuel types were reprocessed during INTEC operations: Al-clad fuel, Zr-clad fuel, stainless steel-clad fuel, and graphite matrix fuel. The dissolution processes varied by fuel type: nitric acid was used for Al-clad fuel, hydrofluoric acid was used for Zr-clad fuel, sulfuric acid/nitric acid and electrolytic/nitric acid were used for stainless steel-clad fuel, and graphite matrix fuel was burned and the ash dissolved in hydrofluoric acid (Loos 2004).

h. The second- and third-cycle extractions were liquid extraction systems that operated in series to provide uranium purification of the first-cycle extraction uranium product. Because they operated on concentrated feed, solutions compared to first-, second-, and third-cycle extractions produced less waste per gallon of feed. The radioactivity level of the second- and third-cycle waste stream was approximately 1,000 times less than that of the first-cycle waste (Loos 2004).

the first-cycle raffinate. The second/third cycle processed the first-cycle uranium product. The second/third-cycle waste was primarily acidified aluminum nitrate, regardless of the type of fuel that was processed. The chemical composition (not radioactivity) of the second/third-cycle waste was similar to the first-cycle waste from Al-clad fuel (Loos 2004).

Unlike other wastes, the second/third-cycle raffinate did not contain any chemicals of concern to waste storage or treatment, such as F, Cl, and Na. It was not so highly radioactive (compared to first-cycle raffinate) that it required cooling or other special considerations beyond that of first-cycle raffinate. Consequently, names reflecting the terms second- or third-cycle waste faded from use. Instead, the names of the waste came from the dominant waste type in terms of waste volume or chemical/radionuclide significance with which it was blended. For example, if 3,000 gal of third-cycle waste was added to 100,000 gal of SBW, the resulting mixture was commonly called SBW, with no reference to the third-cycle portion of the waste (Loos 2004).

The aqueous intermediate liquid waste in Figure 8 from second- and third-cycle extraction was approximately 1% of the initial reprocessing inventory of radioactive material.

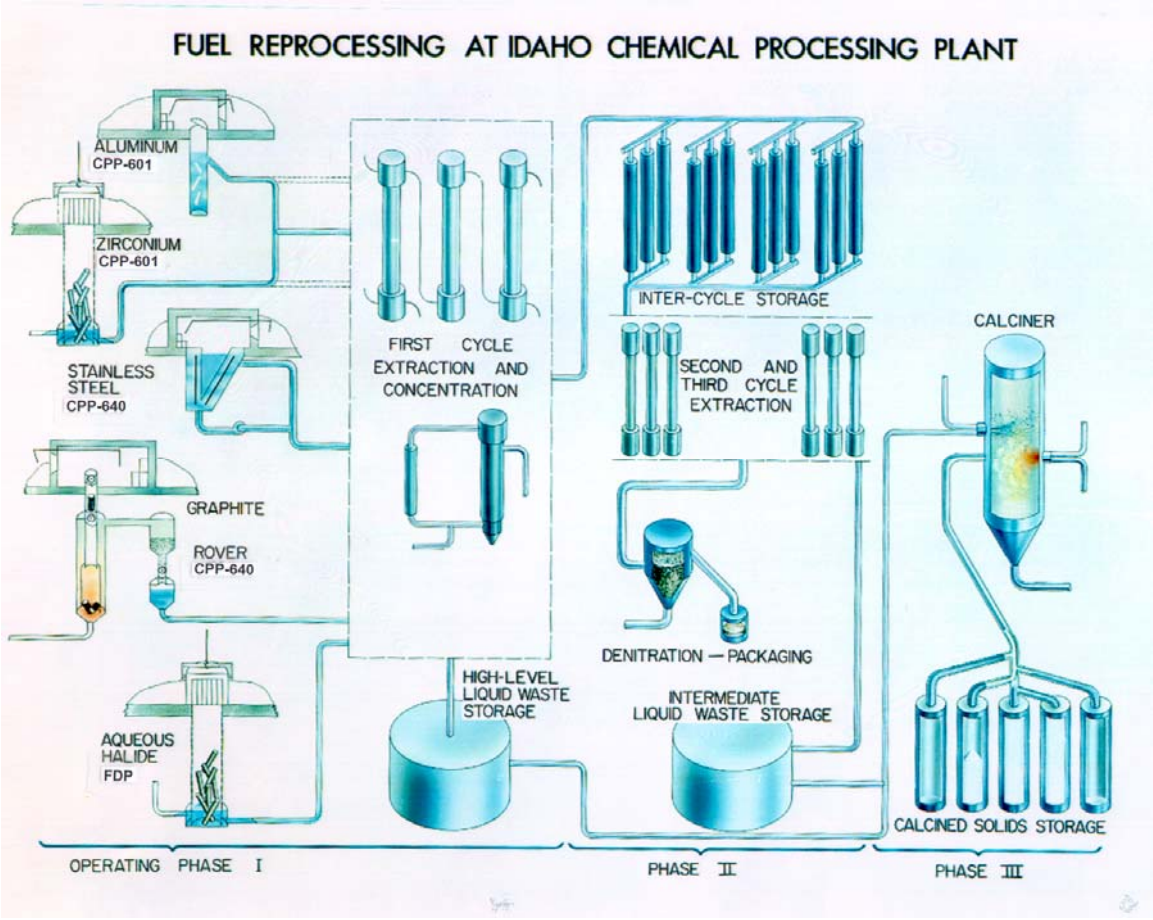

Figure 8. Spent fuel reprocessing at INTEC.

Much of the INTEC reprocessing equipment was designed for contact maintenance rather than remote maintenance. Because INTEC personnel had to access equipment for maintenance purposes frequently, efforts to decontaminate plant areas and equipment to lower radiation fields and allow safe working conditions generated large amounts of decontamination wastes, which were concentrated by evaporation and sent to the TFF. As of November 2003, the remaining TFF waste inventory contained (by volume) 1% first-cycle waste, 2% second-cycle waste, and 4% third-cycle waste. The remaining waste is composed of decontamination wastes and bottoms from the evaporator[i] (Loos 2004).

In general, TFF wastes were managed by segregating the first-cycle solvent extraction waste from other reprocessing wastes because of significantly higher radioactivity levels in the first-cycle waste. The DOE decided not to neutralize waste or combine the first-cycle extraction wastes with other reprocessing wastes, as was the standard practice at other DOE facilities (Knecht et al. 1997). Instead, INTEC maintained the waste in its original acidic form using the 300,000- and 30,000-gal stainless steel storage tanks and physically segregated first-cycle extraction wastes. This decision reduced the volume of liquid waste requiring storage.

Waste volume was reduced further by stabilizing the first-cycle solvent extraction waste and most of the second- and third-cycle extraction wastes through calcination. Calcine results from heating a substance to a high temperature that is below its melting or fusing point. At the INL Site, calcination was carried out in the calciner in the New Waste Calcining Facility where liquid HLW and mixed transuranic waste/SBW are converted into the granular solid known as calcine. The liquid waste is drawn from TFF and sprayed into a vessel containing an air-fluidized bed of granular solids. The bed is heated by combustion of a mixture of kerosene and oxygen. All of the liquid evaporates, while radioactive fission products adhere to the granular bed material in the vessel. The gases from the reaction vessel (called off-gases) are processed in the off-gas cleanup system before they are released to the environment. Calcination reduces the volume of the radioactive liquid waste (usually two to 10 times), so less storage space is needed. The final waste form is a dense powder similar in consistency to powdered detergent. These calcined solids are transferred to the Calcined Solids Storage Facilities, commonly referred to as bin sets. The bin sets are a series of concrete vaults, each containing three to seven stainless steel storage bins (DOE 2002).

Calcined waste is not part of this draft 3116 Determination. The New Waste Calcining Facility, the evaporators, the reprocessing facility, or other nearby facilities are also not included in this draft 3116 Determination.

By evaporating and calcining, and not neutralizing, the liquid radioactive waste, INTEC avoided the construction of up to 195 additional 300,000-gal storage tanks (Knecht et al. 1997). The solidified (calcined) reprocessing waste was transferred to stainless steel bins for storage. By February 1998, the liquid first-cycle extraction waste was removed from the TFF. Only small (1,000–15,000 gal) heels in eight of the 11 300,000-gal storage tanks remained, which could not be removed with existing equipment. Reuse of the first-cycle waste storage tanks to store SBW has resulted in the mingling of the first-cycle waste heels with SBW.

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i. Evaporators have been used at INTEC to reduce the volumes of radioactive waste. Bottoms are the concentrated material remaining after the liquid portion of the waste is evaporated.

Between 2002 and 2005, TFF tank contents were evaporated to less than one million gallons and consolidated into three 300,000-gal tanks. This SBW remains in storage in Tanks WM-187, WM-188, and WM-189. The current SBW inventory is a mixture of wastes from various sources, including:

- Decontamination solutions from past spent fuel reprocessing maintenance activities[j]
- Solids and liquids from the bottom of the TFF tanks, which include some first-cycle waste[k]
- Liquid wastes from ongoing maintenance and closure activities at INTEC
- Second- and third-cycle spent fuel reprocessing extraction wastes.

The DOE is pursuing methods for treating and disposing the remaining SBW, and the three tanks storing SBW will be emptied during the treatment process. Tank heels will be removed as part of closure activities, and they will be treated with the bulk SBW. It is planned to have the treatment of the SBW completed by 2012. As explained previously, the treated SBW is not part of this draft 3116 Determination.

Each of the 11 300,000-gal tanks in the INTEC TFF has a slightly different configuration, waste storage, and processing history (INEEL 1999a, 2000a). The following is a history of the TFF tanks, including the sources of the solid and liquid waste residuals in the tank heel, and a description of the tank contents and level as of April 30, 2005. Liquid levels in the cleaned tanks (WM-180 through WM-186) are not the levels that will remain at closure. The liquid after sampling is left at a depth of approximately 13 cm (5 in.) to avoid the tank drying prior to grouting and creating potential airborne contamination problems during the grouting operations. The liquid will be removed to approximately 3 cm (1 in.) prior to the addition of grout during stabilization operations. The liquid that will be removed will be sent to the SBW treatment system or to operational tanks.

The steam jets have been lowered to 0.97 cm (0.38 in.); therefore, based on steam jet performance, it is conservative to assume that the tank liquid will be lowered to a 3-cm (1-in.) depth (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g). The levels shown in the tank descriptions below are the tank levels at the time the samples were collected after tank cleaning. After the liquid level is reduced using the steam jets, additional liquid will be removed by the grout placements. No credit is taken in the PA and in this draft 3116 Determination for liquid removed during grout placements, which constitutes approximately 3 Ci (per tank), as identified in Table 1. Details of the grout composition are shown in

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j. Typical decontamination cycles from reprocessing fuels consisted of a strong caustic solution to break down the resistant oxide layer, followed by corrosive and/or chelating agents to expose base metal and contaminants. The cycle was completed with a 6-*M* nitric acid flush to restore the oxide layer (INEEL 2002a).

k. Particle size distribution analysis and settling rate testing of the WM-182 and WM-183 TFF heel slurry samples (samples were collected prior to tank cleaning) were performed. The conclusions and recommendations based on the results of this work follow that provide insight into the properties of the solid residual, which include the following (INEEL 2000b):

- The overall results for the standards testing were satisfactory and demonstrated excellent repeatability and acceptable accuracy during the actual TFF sample testing.
- Particle size distribution analysis showed that for both the WM-182 and WM-183 samples, the particles range approximately from a minimum of 0.5 μm to a maximum of 230 μm, with about 90% volume between approximately 2 and 133 μm. The results of the particle size determination for samples from WM-182 and WM-183 are quite similar considering the minute quantities used from the two separate vessels to obtain these results.
- Settling rate testing results were fairly consistent in that it appears that most of the mass of solids settle to an easily redispersed layer.

Appendix C. Information about the grout sequence and the engineering grout pour are included in Section 6. (This liquid removal during grouting activities is not considered to be part of the residual inventory.) The solid residuals exist as fine particles. (All tanks contain cooling coils unless otherwise noted.) Figure 9 summarizes the tanks' volumes as of April 30, 2005.

- Tank WM-180—This tank was placed into service in 1954 and was used to store Al fuel reprocessing wastes. The majority of the first-cycle extraction process waste was calcined in 1966–1967. Since 1972, the tank has been used solely to store SBW. The tank is housed in an octagonal, poured-in-place reinforced concrete vault. This tank does not have an underlying sandpad. On November 30, 2003, WM-180 was in service and contained 276,000 gal of waste. The waste was primarily Process Equipment Waste Evaporator (PEWE) bottoms with small amounts of second- and third-cycle raffinate. The tank was emptied to its heel in 2004 and the waste transferred to the evaporator or to WM-187 (Loos 2004). Tank cleaning operations were completed in October 2004. As of April 30, 2005, Tank WM-180 contains 7,600 gal of liquid after tank cleaning operations and an estimated 540 kg of solid residual (Portage 2005a). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-180 is approximately 1,047 Ci (Portage 2005a).

- Tank WM-181—This tank was placed into service in 1953 and was used only to store SBW. The tank is housed in an octagonal, poured-in-place reinforced concrete vault. This tank does not contain cooling coils or have an underlying sandpad. Tank WM-181 was "emptied" of waste (the volume in the tank was reduced to its heel, using the installed waste transfer equipment) in November 2001. On November 30, 2003, Tank WM-181 contained 22,800 gal of waste. The waste was primarily PEWE bottoms from Tank WL-101 and "other" waste with a small amount of third-cycle raffinate. The tank was never used for storage of first-cycle raffinate (Loos 2004). Tank cleaning operations were completed in May 2004. As of April 30, 2005, Tank WM-181 contains 7,300 gal of liquid after tank cleaning operations and an estimated 250 kg of solid residual (Portage 2005b). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-181 is approximately 475 Ci (Portage 2005b).

- Tank WM-182—This tank was placed into service in 1956 and was used to store Al and Zr fuel reprocessing wastes. The tank was emptied to heel level in 1993 and has since been flushed with small quantities (about 10,000 gal) of SBW. The tank is housed in an octagonal, pillar-and-panel concrete vault. Between December 1996 and June 1999, two transfers of waste were made (15,100 gal total) out of the tank. These transfers removed the tank to heel level. Tank inspection began in 1999. Tank cleaning was completed in September 2002. As of April 30, 2005, Tank WM-182 contains 6,500 gal of liquid after tank cleaning operations and an estimated 1,240 kg of solid residual (Portage 2005c). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-182 is approximately 2,393 Ci (Portage 2005c).

- Tank WM-183—This tank was placed into service in 1958 and was used to store Al and stainless steel fuel reprocessing wastes. The first-cycle extraction process waste was transferred from the tank in 1981, and the tank was refilled with various types of SBW. The tank is housed in an octagonal, pillar-and-panel concrete vault. Between February 1997 and June 1999, 16,000 gal of low-activity waste was sent to WM-183. The accumulated waste in WM-183 was transferred to WM-187 in December 1999, leaving a heel of about 13,000 gal. Tank cleaning was completed in March 2003. As of April 30, 2005, Tank WM-183 contains 8,000 gal of liquid after tank cleaning operations and an estimated 700 kg of solid residual (Portage 2005d). The remaining liquid is water

from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-183 is approximately 1,360 Ci (Portage 2005d).

In 2004, while consolidating SBW to Tank WM-187 from Tank WM-180, it was noticed that the liquid level of Tank WM-183 had increased. Upon further investigation it was determined that approximately 200 gal of SBW and water had leaked into the tank during transfers of WM-180 waste to the evaporator tank system, then to WM-187 during the period between July and October 2004. The leakage was determined to have been through an improper valve setting. The isolation of the tank has been confirmed. The tank has been re-cleaned and sampled. Visual inspection of the tank indicates that the 200 gal of waste that leaked into the tank has been removed and no additional solids are present in the tank. When data from sampling activities are available, the estimated inventory will be confirmed.

- Tank WM-184—This tank was placed into service in 1958 and was used only to store SBW. The tank is housed in an octagonal, pillar-and-panel concrete vault. This tank does not contain cooling coils. Between May and December 2001, WM-184 was emptied to a heel volume of 5,100 gal. After this time, no new wastes were added to the tank. Tank cleaning was completed in December 2003. As of April 30, 2005, Tank WM-184 contains 3,100 gal of liquid after tank cleaning operations and an estimated 560 kg of solid residual (Portage 2005e). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-184 is approximately 1,077 Ci (Portage 2005e).

- Tank WM-185—This tank was placed into service in 1959 and was used to store Al and Zr fuel reprocessing wastes. The first-cycle extraction process waste was calcined in 1983, and the tank was refilled with SBW. The tank is housed in an octagonal, pillar-and-panel concrete vault. The waste was removed from WM-185 in January 2002, leaving a heel of about 13,000 gal of waste in the tank. Tank cleaning operations were completed in October 2003. As of April 30, 2005, Tank WM-185 contains 5,800 gal of liquid after tank cleaning operations and an estimated 720 kg of solid residual (Portage 2005f). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-185 is approximately 1,391 Ci (Portage 2005f).

- Tank WM-186—This tank was placed into service in 1962 and was used to store Al fuel reprocessing waste. The first-cycle extraction process waste was transferred from the tank in 1967, after which the tank was used solely to store SBW. The tank is housed in an octagonal, pillar-and-panel concrete vault. This tank does not contain cooling coils. In May 2001, the tank was emptied to a heel of 20,300 gal. Tank cleaning operations were completed in November 2003. As of April 30, 2005, Tank WM-186 contains 6,600 gal of liquid after tank cleaning operations and an estimated 340 kg of solid residual (Portage 2005g). The remaining liquid is water from tank cleaning. Prior to grouting, the remaining liquid volume will be removed to a 3-cm (1-in.) depth or approximately 1,000 gal. The estimated inventory at closure for Tank WM-186 is approximately 646 Ci (Portage 2005g).

- Tank WM-187—Placed into service in 1959, Tank WM-187 was used to store Al and Zr fuel reprocessing wastes. The first-cycle extraction process waste was calcined between 1990 and 1993, and then SBW was added to the heel. The tank is housed in a square, reinforced-concrete vault. This tank is being used to collect SBW solutions from other TFF tanks undergoing cleaning activities prior to final closure (Loos 2004). As of April 30, 2005, Tank WM-187 contains 271,500 gal of liquid, which was transferred from cleaned tanks (ICP 2005a). This tank will be

cleaned to remove highly radioactive radionuclides to the maximum extent practical and meet the other criteria in Section 3116 of the NDAA.

- Tank WM-188—This tank was placed into service in 1959 and was used to store Al and Zr fuel reprocessing wastes. The tank was emptied to heel level when the waste was calcined in 1998. The tank is housed in a square, reinforced-concrete vault (Loos 2004). As of April 30, 2005, Tank WM-188 contains 282,600 gal of liquid that is primarily SBW (ICP 2005a). This tank will be cleaned to remove highly radioactive radionuclides to the maximum extent practical and meet the other criteria in Section 3116 of the NDAA.

- Tank WM-189—Placed into service in 1966, Tank WM-189 was used to store Zr fuel reprocessing wastes. The first-cycle extraction process waste was emptied to heel level in 1996. Since then, the tank has been used to store a variety of SBW solutions from continuing operations (high-fluoride decontamination wastes, bottoms from the evaporator tank system, and other SBW). The tank is housed in a square reinforced concrete vault (Loos 2004). As of April 30, 2005, Tank WM-189 contains 282,300 gal of liquid that is primarily SBW (ICP 2005a). This tank will be cleaned to remove highly radioactive radionuclides to the maximum extent practical and meet the other criteria in Section 3116 of the NDAA.

- Tank WM-190—Tank WM-190 was designated as a spare tank for use in emergencies and was never placed into service. This tank was never used to store any waste, including first-cycle extraction process waste, although the tank has been contaminated with this waste. By 1980, approximately 7,000 gal of liquid had accumulated in the tank from two sources: (1) rainwater that collected in the vault sump was jetted into the tank and (2) a small quantity (less than 50 gal total) of first-cycle extraction process waste was passed inadvertently through the transfer valve. The transfer valve was opened slightly and closed (to confirm valve closure) prior to starting waste transfers to tanks that shared a common transfer line (Loos 2004). (Waste from the last transfer would collect above the valve.) In 1982, the tank contents were transferred using a specially designed sump pump, which left approximately 500 gal of liquid in the tank as of April 30, 2005 (ICP 2005a). This tank will be cleaned to remove highly radioactive radionuclides to the maximum extent practical and meet the other criteria in Section 3116 of the NDAA.

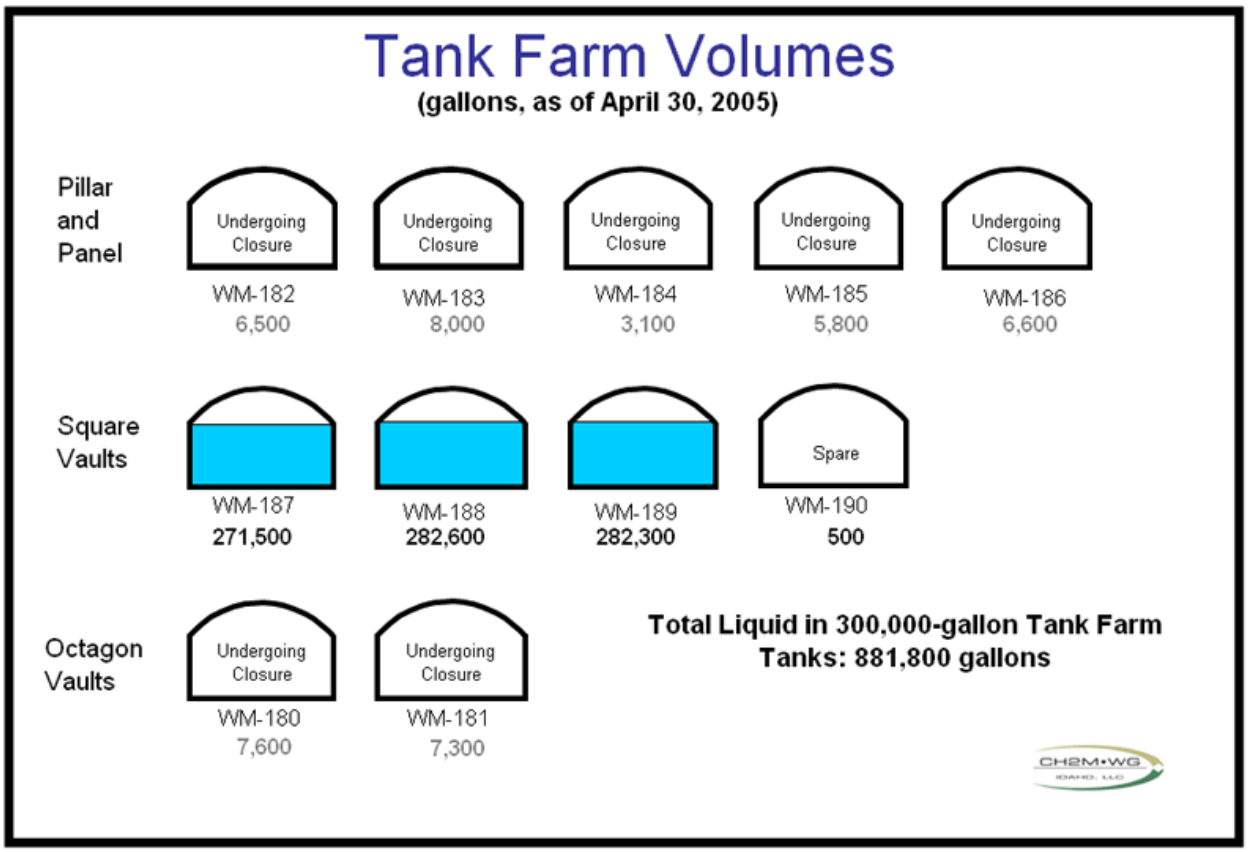

Figure 9. Volumes in TFF 300,000-gal tanks as of April 30, 2005 (ICP 2005a).

The four 30,000-gal tanks (WM-103 through WM-106) were placed into service in 1956 and were used primarily (the tanks held injection well waste and process evaporator condensate for a short period in 1983 and all of the 30,000-gal tanks held steam valve condensate after 1990) for first-cycle extraction process waste from stainless steel and Zr fuel reprocessing. Each of the tanks were emptied to heel level (8–13 cm [3–5 in.] in the tank bottom, approximately 1,000 gal per tank)[l] in 1974 and flushed with water. They remained at heel level until 1982, when they were used to store condensate liquid (not first-cycle process wastes) from the PEWE for 5 months, after which they were emptied again to heel level and taken out of service in February 1983.[m] The waste inlets to these tanks were subsequently cut and capped; however, the outlets remained operational to permit tank decontamination activities to occur. In 1990, the 30,000-gal tank heels were sampled and analyzed. The results indicated a small amount of radioactivity

l. Jacobson, V. L., INEEL, Memo to File: "Discussion on WM-103 through -106 with Frank Ward, BBWI Operations Engineering," September 26, 2001.

m. The PEWE treats diluted, low-activity waste by boiling the waste, and condensing and collecting the vapors. The PEWE splits the waste into two streams. One stream is a small volume of concentrated liquid that contains most of the chemical and radioactive constituents that are originally in the PEWE feed. The concentrated waste stream is called the "bottoms." Typically, the PEWE generates 1–2 gal of bottoms from every 100 gal of feed, concentrating the waste feed by a factor of 50–100. The second waste stream is the condensed vapor that left the PEWE. This stream is called "process condensate," which is a large volume of waste that contained 98–99 gal of every 100 gal of PEWE feed solution. Most of the chemical and radioactive constituents in the PEWE feed (such as Al, Zr, ^{137}Cs, and ^{90}Sr) are nonvolatile and concentrated in the evaporator bottoms. As a result, the PEWE condensate is relatively clean water, containing only trace quantities of most chemicals and radionuclides (Loos 2004).

(the ^{137}Cs concentrations ranged from 1.2E–07 to 1.1E–05 Ci/L) (DOE-ID 2003b). Then the tanks were each flushed with 12,000 gal of water and emptied to their heels (DOE Idaho 2004a).

In April 2004, the 30,000-gal tanks were cleaned by flushing each tank with 3,000–5,000 gal of water and flushing the piping three times. (The length of pipe being cleaned is flushed with three volumes of water. The volume of water used is equal to the volume of the length of pipe.) The tanks were inspected after flushing and the samples were analyzed to determine the concentrations of residual radionuclides. A conservative estimate of the solid residual was made based on the presence of a film layer in the lower half of the tank. Approximately 19 kg of waste residual remain in each tank is estimated based on video inspection and sampling and analysis (Portage 2005h). Each 30,000-gal tank has an estimated inventory at closure of approximately 36 Ci.

2.3 Tank Farm Facility Closure Activities and Status

The DOE is closing the TFF tanks in response to a January 1990 Notice of Noncompliance and subsequent Consent Order (State of Idaho 1992). The Idaho Department of Health and Welfare[n] and EPA issued the Notice of Noncompliance to the DOE because the tanks in the TFF did not meet the secondary containment requirements as set forth by Idaho Administrative Procedures Act (IDAPA) 58.01.05.009 (40 CFR 265.193). The resulting 1992 Consent Order (and subsequent modifications) (State of Idaho 1992, 1994, 1998, 1999) required the DOE to permanently cease use[o] of the five 300,000-gal tanks that are contained in five pillar-and-panel vaults by June 30, 2003. The Consent Order also required the DOE to permanently cease use of the remaining 300,000-gal tanks by December 31, 2012, or bring the tanks into compliance with secondary containment requirements. The DOE decided to close the TFF tanks because radiation fields would make compliance with secondary containment requirements impractical, and because the DOE did not anticipate a need for such storage after 2012. (Compliance is impractical because the radiation fields in the tank vault would prevent practical entrance by personnel to add equipment or upgrades to the vaults to meet the secondary containment requirements [DOE 2002].)

In addition, in October 1995, the State of Idaho, the Department of the Navy, and the DOE settled the cases of *Public Service Co. of Colorado* v. *Batt,* No. CV 91-0035-S-EJL (D.Id.) and *United States* v. *Batt,* No. CV-91-0065-S-EJL (D.Id.) (State of Idaho 1995). Among other things, the resultant settlement agreement required the DOE to complete the process of calcining all remaining non-sodium-bearing liquid HLW by June 30, 1998, and to remove SBW from the INTEC TFF tanks by December 31, 2012. In accordance with this settlement agreement, the DOE completed the removal of first-cycle extraction waste from the TFF tanks to heel[p] level in February 1998. The tanks were then used to store additional SBW. The DOE has evaluated options for waste treatment and INTEC facility disposition. This evaluation is discussed in the *Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement* (DOE 2002). After a record of decision for the final environmental impact statement is issued, the DOE will finalize plans for treating the SBW.

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n. On July 1, 2000, the Division of Environmental Quality, a division within the Idaho Department of Health and Welfare, was elevated to the State of Idaho Department of Environmental Quality (DEQ). This department now oversees the implementation of the Consent Order.

o. "Cease use" means to empty the tanks down to their heels (i.e., the liquid level remaining in each tank after lowering to the greatest extent possible by use of existing transfer equipment). Closure plans developed for these tanks will address the remaining heel and vaults and the use of these tanks and equipment for closure, including any flushing or other cleaning of the tanks (State of Idaho 1998).

p. "Tank heel" means the liquid level remaining in each tank after lowering the level to the greatest extent possible by using existing transfer equipment, such as steam jets.

The TFF is being closed in accordance with DOE requirements as a radioactive waste storage facility, and with HWMA/RCRA requirements for closure of an interim status HWMA/RCRA tank system. These requirements include preparing several documents. The documents describe DOE's actions to close tanks and meet closure objectives. The DOE requires closure plans, a PA, and a composite analysis to address radioactive constituents. To meet HWMA/RCRA requirements, closure plans and sampling and analysis plans (SAPs) addressing the hazardous waste constituents are required. The DOE will not proceed with irreversible closure actions such as grouting until consultation with NRC is complete and the State of Idaho has approved the HWMA/RCRA closure plans.

The following major documents have been prepared for the TFF closure[q,r]:

- Deactivated HLW facility DOE Tier 1 closure plan (DOE-ID 2003c)—Developed in accordance with DOE Order 435.1 (2001) requirements and guidance, this plan defines the approach and plans for TFF closure and includes a PA (DOE-ID 2003b) and a composite analysis (DOE-ID 2003d). The PA documents the projected radiological dose impacts associated with TFF closure to meet the requirements of 10 CFR 61, Subpart C, "Performance Objectives." The composite analysis is a planning tool that provides a reasonable expectation that the proposed closure activities will not result in the need for future corrective or remedial actions to ensure protection of the public and environment. This analysis shows that radiation doses to members of the public through the groundwater pathway at the INL Site boundary are protective of human health.

- HWMA/RCRA closure plans (DOE-ID 2003a, 2003e, 2003f, 2004; DOE Idaho 2004a, 2004b) and associated SAPs (ICP 2004a, 2004b; INEEL 2001, 2002b, 2003).

In general, the closure process includes removing the SBW for treatment then closing the tanks to meet RCRA and Section 3116 criteria. The TFF tank system's closure process includes waste removal; cleaning of the tanks, piping, and ancillary equipment (to remove waste to the maximum extent practical); and stabilization[s] of the tank configuration and ancillary equipment. To complete SBW removal, as much of the remaining liquid and solid waste residue will be removed to the maximum extent practical from the tanks and ancillary equipment. Following waste removal from the tanks and TFF cleaning activities, confirmatory sampling and analysis will be performed to assess the decontamination effectiveness and for waste characterization. (As discussed below, this sampling and analysis has already been performed for the tanks and ancillary equipment that have been cleaned as of May 2005.)

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q. Following closure to these requirements, decisions and actions regarding final capping, monitoring, and long-term maintenance of the site will be conducted as part of the CERCLA program.

r. In December 2002, the DOE prepared a draft waste determination for the TFF ("Idaho Nuclear Technology and Engineering Center Tank Farm Facility Residuals–Waste-Incidental-to-Reprocessing Determination Report," DOE-ID-10777, Draft Rev. B, December 2002) to demonstrate that the residual waste in the TFF tanks met the DOE's requirements (DOE M 435.1-1, 2001) for managing the waste as LLW and submitted the draft document and the PA (DOE-ID 2003b) to the NRC for review. In general, the NRC found that the DOE demonstrated that the waste meets DOE's and NRC's waste determination guidance to manage the waste as LLW. The NRC also concluded that the DOE's determination that the residual waste from tank closure activities can be managed as LLW "has sound technical assumptions, analysis, and conclusions with regard to protecting public health and safety, and the environment" (SECY-03-0079, 2003). However, with the enactment of Section 3116 of the NDAA, the DOE did not formally issue the draft waste determination.

s. Waste stabilization is intended to minimize radiation exposures to the public caused by leaching of radionuclides and chemicals out of the waste form and by intrusion into the waste. Stability limits exposures to an inadvertent intruder by providing greater assurance that the waste form will remain recognizable and nondispersible.

Some residuals that cannot be removed by the cleaning process or other technically practical means will remain. The waste residuals will be sampled and analyzed to determine the concentrations of radionuclide and hazardous constituents remaining in the tanks. After analytical data show that performance objectives are met for the entire TFF tank system (i.e., after all tanks and ancillary equipment have been cleaned), the DOE plans to stabilize the tank system by filling the system with grout. Process lines will be decontaminated and capped, and all lines (including process lines) that provide a pathway to the tanks will be grouted and capped (Appendix C; DOE-ID 2003c).

This draft 3116 Determination includes the evaluation of all TFF equipment and structures that are potentially contaminated with reprocessing wastes as a result of past INTEC reprocessing operations. These structures, systems, and components (SSCs) will be isolated and grouted as a part of the INTEC TFF final closure and include the stainless steel tanks, concrete vaults, sandpads, piping, encasements, valve boxes, and instrumentation lines. The TFF tank cleaning operations began in 2002 for those tanks emptied of SBW. Closure of the INTEC TFF is being conducted in phases to support continued INTEC operations.[t] As of May 2005, Tanks WM-180 through WM-186 and WM-103 through WM-106 have been cleaned along with the ancillary equipment of each tank. Samples have been collected from the tanks and ancillary equipment, and the results have been reported in data quality assessment (DQA) reports (ICP 2004c, 2004d, 2004e, 2004f, 2004g, 2004h, 2004i, 2004j, 2005b; INEEL 2004a, 2004b). Waste liquids and solids removed and generated during tank closure activities are being collected for final treatment and disposal.

2.3.1 Cleaning Approach for Stainless Steel Tanks and Vaults

A tank cleaning system comprising a washball, directional nozzle, and modified steam-jet pumping system has been developed and used successfully thus far in the TFF tank cleaning operations. Figure 10 illustrates this cleaning system. During washball and directional nozzle operations, the steam-jet ejectors are operated to remove the waste-containing slurry from the tank. The goal of tank cleaning is to remove as much waste as practical. During this operation, radiation levels are monitored on the steam-jet transport line as an indication of cleaning effectiveness. Monitoring the radiation levels near the transport line provides the cleaning system operators and project manager an indication of when continued tank cleaning ceases to be effective. When radiation levels decrease to the lowest value and remains constant, cleaning is stopped and the tanks are inspected. If visual inspection via a remote-controlled camera confirms that the tank has been cleaned to the extent practical, then samples are collected and analyzed to verify performance objectives are met. During the visual inspection, residual solid waste depths are estimated by comparing the solids depths to benchmarks within the tanks such as cooling coil support brackets and associated welds. For example, in tanks with cooling coils, the bottom weld and stainless steel bracket thickness measures 0.97 cm (0.38 in.). Knowing this thickness, the depth of waste next to these brackets can be estimated. A reflection from the stainless steel at the tank bottom indicates that no solids are present (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g). The radiation monitor allows tank cleaning to proceed without repeated visual inspection or sample collection and aids in ensuring that as much waste as practical is removed from the tanks (Kimmitt 2002). Samples of the residual are collected with small positive-displacement pumps. Submersible pumps are lowered into the bottom of tanks or vaults through risers. The pump is activated and liquid and solids are pumped to

t. INTEC operations include management of SBW and newly generated liquid waste. Also included are the evaporator tank system and the PEWE (the PEWE is not part of the closure or this draft 3116 Determination). Some of the newly generated liquid waste is the water from tank cleaning operations. This liquid is sent to Tank WM-187 through existing process piping and valve boxes. Additionally, these lines and valve boxes are used to transfer the liquid to the PEWE. A phased approach to cleaning was used at TFF because the transfer of liquid waste was necessary, closure of tanks and associated piping had to be scheduled to allow liquids to be transferred from a tank being cleaned yet not restrict operations such that another tank cannot be cleaned, or valves and piping that allow waste to be transferred to the PEWE are not closed prematurely.

sample containers on the surface. The submersible pump can only reach the residual directly beneath the riser through which it is lowered. However, because the residual will be agitated before sampling, it is reasonable to assume that the liquid in the tank is homogeneous and the flocculent solids may be in suspension.

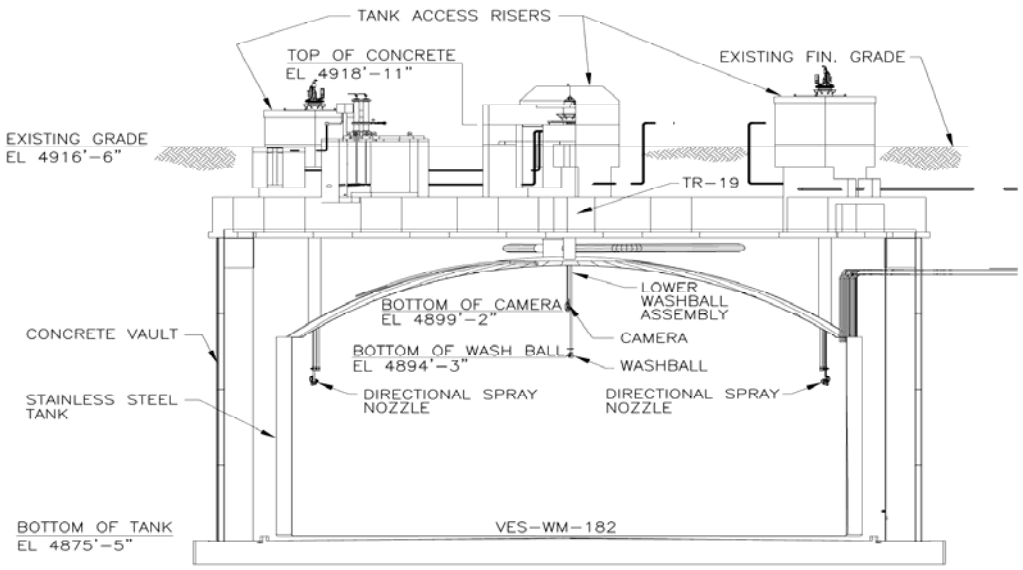

Figure 10. Typical tank cleaning system.

Prior to its implementation, the TFF tank cleaning system was tested in a full-scale mockup tank (INEEL 1999b) using simulated waste (EDF-015722-041, 2000). The system demonstrated effectiveness in subsequent tank cleaning. The washball/directional nozzle tank cleaning system and the modified steam-jet pumping system were used to slurry the solid and liquid wastes and remove them from the tanks. Steam jets were modified by cutting the steam supply and discharge lines and installing a new steam jet lower in the tank. During cleaning system development, the INL Site and the Tanks Focus Area (TFA) performed a review of tank cleaning technologies. The DOE formed the TFA to address all aspects of remediating radioactive wastes from underground storage tanks DOE-wide, including tank cleaning technology. The TFA review is described in a report prepared by the Pacific Northwest National Laboratory (PNNL 2001). This review focused not only on the technical feasibility and appropriateness of the approach selected by the INL Site but also on technology gaps that could be addressed by using technologies or performance data available at other DOE sites and in the private sector. The review supported the design and implementation of the INL Site cleaning system. As a result of this development and testing work, the cleaning system (washball, directional nozzle, and modified steam-jet pumping system) has performed beyond expectations as demonstrated by a comparison of the total post-decontamination inventories estimated in the PA and calculated for WM-182, as presented in Appendix A.

The tank vaults are cleaned by iterative flushing with water. The water is removed using the existing steam jets. Access to the vaults is limited and initial radionuclide inventories are considerably less than the tanks (DOE-ID 2003b); therefore, flushing is a practical and effective cleaning method. Process piping in the TFF is cleaned by triple flushing, which consists of flushing three piping system volumes through the system with a pressure equal to previous waste transfers to ensure that the pipe area contacted by waste has been contacted by water and rinsed during the flushing operations; this method has been shown to be effective based on analytical data.[u] The acidic nature of the waste and the procedures that required flushing with water after waste transfers during operations limited accumulation of residual in piping that needed to be removed.

The vaults provide secondary containment for tank leakage. The tanks have not leaked during the life of the TFF. The contamination in the vaults for Tanks WM-185 and WM-187 (which resulted from back-siphoning events as discussed in Subsection 2.4.4) is considered in the PA. In all tank vaults, rainwater/snowmelt in-leakage through the vault roof has been pumped periodically from the vault sumps to waste tanks.

Tank waste residuals remaining after cleaning and before grouting consist of a relatively small amount of solids (see Subsection 2.2) and contaminated flush water. Extensive mockup testing shows that most of the remaining flush water and some solids will be removed during the grouting process for stabilizing the residuals. This action will be accomplished by using the grout to push and corral the waste residuals toward the removal equipment (jet pumps). Any remaining residual liquid will be stabilized with a grout material. Prior to grouting, the small amount of liquid waste in the vault sumps will be emptied using the existing steam-jet pumps. The lines connecting the vault sumps to the tanks will be grouted, followed by grouting of the vaults.

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u. Analysis of residual metal contamination in process waste piping was performed in 2002. The estimate of residual radionuclide inventory in piping is based on this sampling and analysis. The *Sampling and Analysis Plan for the Post-Decontamination Characterization of the Process Waste Lines from INTEC Tank Farm Facility Tanks WM-182 and WM-183* (INEEL 2001) was prepared to define the steps to collect data that represent residual in piping. The process waste lines in the WM-182 and WM-183 tank systems have carried acidic waste in solution and have routinely been flushed after waste transfer with either acid or an acid and water flush. During closure of the tank systems, the piping was triple rinsed with water to remove loose residual waste. Sections of horizontal and vertical process waste line have been removed from Tank WM-182. Samples from the decontaminated process waste lines will be collected and the data will be used to represent the effectiveness of triple rinsing all of the lines remaining in the WM-182 and WM-183 tank systems. The pipe to be sampled was removed from the system. The piping was filled with water, allowed to equilibrate for a minimum of 30 minutes, and sampled and analyzed for total metals.

The concentration of metals in liquid (mg/L) SBW is proportional to the concentrations of radionuclides (mg/L). Therefore, the residual metal concentrations in piping can be used to calculate the residual radionuclide concentrations. The maximum concentration of each metal result in rinsate samples of piping was summed with the other 23 metals analyses. This yielded a concentration of 2.9 mg/L. The rinsate was collected from 46-cm (18-in.) lengths of the 6-cm (2-in.) diameter pipe. It is not assumed that the chemical properties of radionuclides and metals are the same. Therefore, the conservative assumptions shown below were used to ensure the decontamination factor for metals was not overstated, the fixed radioactive contamination would be included in the estimate, and to provide for the possibility of greater concentrations being found in other piping. This method of estimate radionuclide concentrations in the piping was reviewed previously by the NRC and found to be acceptable (SECY-03-0079, 2003). Several conservative assumptions were made to ensure this estimate is conservative:

- The mass of metals equals the mass contained in 0.3 m (1 ft) of piping. Therefore, the starting value was one-third greater than indicated by the analysis.
- The sample volumes were less than 1 L. The data were not adjusted downward to correspond with the actual sample volumes.
- A safety factor of 500 was applied to the data to ensure a reasonably conservative estimate.

The use of these data results in a mass of 15.5 kg of solids remaining in the piping (DOE-ID 2003b).

2.3.2 Results of Tank and Ancillary Equipment Cleaning from 2002 to 2005

The results of visual inspection and sampling and analysis performed after tank cleaning operations were completed provide evidence that the cleaning technology used for the 300,000-gal tanks, 30,000-gal tanks, and ancillary equipment effectively removes the majority of highly radioactive radionuclides from the tanks while keeping occupational exposure to radiation as low as reasonably achievable (ALARA).

For the seven 300,000-gal tanks that have been cleaned as of May 2005 (Tanks WM-180 through WM-186), the washball and directional nozzle tank cleaning system was used to wash the tank walls and ceiling. The high-pressure water from the washball and directional nozzle also agitated the tank heel to suspend the solids and facilitate heel removal. The washball and directional nozzle were lowered into the tank through one of the tank risers. The water from the washball and nozzles hit the tank walls, roof, and heel, and dislodged the bulk of the contamination on the walls and ceiling of the tank to allow subsequent removal using the steam jets. A camera and lighting system was used to monitor the decontamination and heel removal effectiveness. Existing tank equipment and new equipment used inside a tank waste removal and decontamination operations were left inside that tank after these operations were completed for permanent disposal when practical. However, the washball and directional nozzle were decontaminated when moved to a new tank. The four remaining 300,000-gal tanks (WM-187 through WM-190) will be cleaned using the same methods after the SBW is removed for future treatment.

Visual examination indicated that the water spray removed solids from the tank walls easily. After tank cleaning operations were completed, post-decontamination samples of the tank residuals and from ancillary equipment were obtained. Samples from Tanks WM-180 through WM-186 were collected. A minimum of five samples were collected from each tank. The remaining tank contents were agitated between sample collections to ensure random selection of the samples. Because few solids remained on the tank bottoms, only a few grams of solids from one tank (WM-183) were retrieved during the sampling activities. Attempts to sample solid material in other tanks failed because of a lack of solid material available to collect. Figures 11 and 12 are photographs of the interior of WM-183 before tank cleaning and after tank cleaning, respectively. Figure 11 shows the side of the tank and the cooling coil on the tank wall, which is approximately 20 cm (8 in.) above the tank floor. Figure 12 shows the cleaned tank floor and cooling coils, which are obscured by waste in Figure 11. After cleaning, the tank floor has only isolated areas of solid residual as shown in the photograph (see Figure 12). The cooling coil support brackets and welds on the floor and on the cooling coil base plate (0.97 cm [0.38 in.]) are clearly visible. Similar results were achieved for the other cleaned tanks. Post-decontamination photographs of the cleaned tanks are presented in Appendix A.

The four 30,000-gal tanks (WM-103 through WM-106) were flushed with water to remove any residual waste. A tank cleaning system was not used for these tanks because of the low radionuclide concentrations remaining after previous tank flushing (see Subsection 2.2). Post-decontamination samples (at least five from each tank) from Tanks WM-103 through WM-106 were also collected (Portage 2005h).

Figure 11. Photographs of the interior of TFF Tank WM-183 before tank cleaning.

Figure 12. Photograph of the interior of TFF Tank WM-183 after tank cleaning.

Sampling and analysis were performed in accordance with SAPs (ICP 2004a, 2004b; INEEL 2001, 2002b, 2003) prepared in conjunction with the tank and ancillary equipment closure plans. Each SAP used the data quality objective (DQO) process to determine the sampling strategy, number of samples, and analytical methods to be used. Issues such as the representativeness of the samples obtained and the homogeneity of the population sampled are also addressed. The DQO process is a planning approach developed by the EPA for use in preparing sampling designs for data collection activities that support decision-making. The process is used to ensure that the type, quantity, and quality of data used in decision-making are appropriate for the intended application (EPA 2000a). The analytical results have been reported in a series of DQA reports (ICP 2004c, 2004d, 2004e, 2004f, 2004g, 2004h, 2004i, 2004j, 2005b; INEEL 2004a, 2004b). A DQA is a scientific and statistical evaluation of the quality of the collected characterization data to determine whether the data can be used to meet the DQOs that were established (EPA 2000b).

In this case, the results of post-decontamination sampling and analysis were used to determine the concentrations of the radioactive and hazardous constituents remaining in the tanks and ancillary equipment. Analysis results were used to confirm that the radionuclide concentrations met the closure requirements, and that they were bounded by the concentrations assumed in the conservative inventory in the 2003 PA (DOE-ID 2003b). The results were also used to estimate the residual TFF radionuclide inventories. Results of sampling and analysis of the waste residuals indicate that the radionuclide inventory in the tanks after cleaning is an order of magnitude lower than that estimated in the 2003 PA. A comparison of the PA conservative inventory and the Tank-WM-182 inventory at closure is presented in Appendix A.

2.4 Residual Waste Inventory in Tank Farm Facility Structures, Systems, and Components

Although the tank and ancillary equipment cleaning activities have been shown to be successful, some waste residuals will remain in the TFF. These residuals contain radionuclides that will potentially create a radiation dose to members of the public. The residual waste inventory at closure, an estimate of

the amount of residual waste remaining after the TFF tank system is closed, is used in this draft 3116 Determination to evaluate this potential radiation dose and to support demonstration of compliance with Section 3116 criteria.

The residual waste inventory at closure is based on validated analytical data from tanks sampled between 2002 and 2005, after cleaning operations. (See Subsection 2.3.2 for a description of and references to the SAPs and DQA reports.) This inventory describes the radionuclide concentrations in the TFF tank system residuals, assuming that the residuals in the four 300,000-gal tanks and ancillary equipment remaining to be cleaned have a radionuclide inventory similar to that of the residuals in the seven tanks and ancillary equipment that have already been cleaned. Radionuclide concentrations are decayed to 2012, the year of final TFF closure. This date is used because performance objectives will not be met until all tanks and ancillary equipment are cleaned and stabilized with grout. (However, this inventory does not include the grout, nor is the inventory reduced by the amount of residuals that may be further removed by grouting operations.) The following subsections describe the residual waste inventory at closure for the 300,000-gal tanks, 30,000-gal tanks, sandpads, and piping; how each inventory was developed; and how the inventory will be used to support the demonstration of compliance with Section 3116 criteria.

2.4.1 Models and Calculations Used to Develop Residual Waste Inventory at Closure

Historically, the INTEC uses ^{137}Cs analytical data to aid in process control and estimate radionuclide inventories. This radionuclide is a major component in INTEC waste streams and is a strong gamma-emitting radionuclide, which makes it easy to detect with confidence in radioactive waste streams. Because no or limited analytical data are available for many other radionuclides, ^{137}Cs data are also used to estimate values for other radionuclides. Radionuclide inventories for all tanks are presented in Appendix A. The ^{137}Cs sample data and ORIGEN2 (Croff 1980) and analytical ratios, as explained below, are used to estimate radionuclide inventories when analytical data for specific radionuclides are not available.

The gamma-emitting radionuclide ^{137}Ba contributes heavily to the overall activity in the inventory. This activity level has a tendency to "overshadow" the activity of lower energy or lower activity radionuclides (e.g., ^{14}C). This occurs because the intensity and high concentration of the gamma-emitting radionuclides create overlap, scatter, and other interferences during analysis, which in effect, result in a loss in instrument sensitivity. Depending on the levels, this results in either difficulty or the inability of the laboratory to measure low-energy/low-activity nuclides.

The ORIGEN2 model is used to predict radionuclide inventories based on nuclear fuel types (e.g., Al, Zr, and stainless steel) that have been processed at INTEC. Input parameters to the model are adjusted to allow the model to predict SBW radionuclide inventories based on weighted averages of the different fuel types that have been processed. The INL Site uses radionuclide and ^{137}Cs values from the ORIGEN2 model to calculate a radionuclide-to-^{137}Cs ratio called the ORIGEN2 ratio for each radionuclide. Analytical results from SBW samples that contain reliable data for a specific radionuclide and ^{137}Cs are used to calculate an analytical ratio. All reliable analytical data are used to generate average analytical ratios for radionuclides in SBW. These two ratios, the ORIGEN2 and analytical ratios, have been modeled and published and are informally called the Wenzel Tables (EDF-FDO-006, 1997). These tables have been integrated into a FORTRAN program, which is used to calculate radionuclide data (when analytical data are lacking) for SBW based on ^{137}Cs sample results (see the following example).

Example of using ORIGEN2 and analytical ratios to calculate radionuclide data for SBW:

- Data from Wenzel tables

- The concentration of ^{238}U is 2.2E–08 Ci/L (radioactive decay to 2012) as calculated by the ORIGEN2 model for SBW

- The ^{137}Cs concentration in 2012 for SBW is 2.8E–02 Ci/L (based on analytical results from historical analysis of SBW)

- ORIGEN2 and analytical ratios

  - The ratio for ^{238}U is calculated by dividing the concentration of ^{238}U by the concentration of ^{137}Cs or 2.2E–08/2.8E–02, yielding an analytical ratio = 8.2E–07 for ^{238}U since it was based on ORIGEN2 results

- Concentrations based on ^{137}Cs analytical results and ORIGEN2 and analytical ratios:

  - Based on the analytical data gained from gamma spectroscopy, the concentration of ^{137}Cs detected in Tank WM-182 is 2.2E–04 Ci/L (radioactive decay to 2012). Therefore, the estimated concentration in 2012 for ^{238}U based on the ^{137}Cs analytical data and the analytical ratio would be 2.2E–04 Ci/L x 8.2E–07 = 1.8E–10 Ci/L.

(Note: The values used in the calculations have been rounded.)

The analytical ratio is used preferentially over the ORIGEN2 ratio to calculate TFF inventories when available, because the analytical ratio is modified to the measured ^{137}Cs concentrations (DOE-ID 2003b). These analytical ratios are used to provide an inventory that has a basis in analytical data. The ORIGEN2 and analytical ratios are also used to validate analytical results of difficult-to-measure isotopes when they are found in waste streams with moderate to high radionuclide concentrations. The results of laboratory analyses are used to modify the ORIGEN2 results to better reflect the concentrations of radionuclides in the inventory.

Expensive and time-consuming separation techniques are available that increase the likelihood of obtaining more reliable laboratory results. However, even when these techniques are applied, a detection of a radionuclide may not be made or may be at concentrations that have a high degree of uncertainty. In lieu of these techniques, the industry has developed and adopted the use of tools such as the ORIGEN2 model to obtain reasonable estimates of low-energy/low-activity components when evaluating large inventories of radionuclides. This practice is widely recognized throughout the industry as a means to obtain reasonable estimates of radionuclides that are difficult or impossible to measure in the presence of high-energy/high-activity components (DOE-ID 2003b).

2.4.2 Residual Waste Inventory at Closure in 300,000-gal Tanks

The residual waste inventory at closure for the 300,000-gal tanks is based on analytical data from samples collected after tank cleaning from 2002 through early 2005. Of the 300,000-gal tanks, Tanks WM-180, WM-181, WM-182, WM-183, WM-184, WM-185, and WM-186, and the associated ancillary equipment have been cleaned. Analytical data from the cleaned tanks are the primary source of information for creating the residual waste inventory at closure. The radionuclide concentrations used to calculate inventories for each tank are based on analytical data decayed to 2012. Concentrations for radionuclides for which samples were not analyzed or for which radionuclides were not detected are replaced with the estimates calculated from the ORIGEN2 and analytical ratios, and analytical results for ^{137}Cs as described in Subsection 2.4.1. Samples were analyzed for all radionuclides that were important dose contributors. Due to the nature of the SBW, some radionuclides were hard to detect without specialized analyses or the development of new analytical methods. These radionuclides were not

analyzed if they were not considered to be significant dose contributors. Therefore, these radionuclide concentrations were estimated as described in Subsection 2.4.1.

The residual solids data are based on a sample of solids collected from Tank WM-183. Although seven 300,000-gal tanks have been cleaned, only one sample of solid material was able to be collected because of the lack of residual solids in the cleaned tanks.

The residual solids inventory for those radionuclides modeled using the ORIGEN2 analytical ratios is based on the mean of the ^{137}Cs concentrations from Tank WM-188 collected prior to cleaning (decayed to 2012). Tank WM-188 ^{137}Cs data are used because this tank had the highest measured ^{137}Cs concentration of any TFF tank before or after cleaning and because only one solid sample could be collected from the tanks after cleaning operations. Therefore, using the higher concentration data from Tank WM-188 is conservative. Concentrations estimated from the analytical ratios are used for radionuclides not able to be detected.

The single solid sample from Tank WM-183 introduces uncertainty to the radionuclide inventory for the TFF tanks. While it is difficult to quantify the uncertainty, a qualitative uncertainty discussion is included in the following paragraphs to show that the inventory for the entire TFF is reasonably conservative. The inventory that is shown in this draft 3116 Determination is based on the data collected from the cleaned tanks. The PA inventory is compared to the residual inventory in Appendix A, Table A-12.

In all of the cleaned 300,000-gal tanks, five liquid samples were collected from three locations, with two of the locations sampled twice. Limited access to the tank through existing risers requires this type of sample collection effort. While all tanks were sampled, only a few grams of solids were collected from Tank WM-183. Many factors contribute to the overall reasonably conservative inventory based on a single sample of solids, which are:

- Tank WM-190 has never been used and has been only slightly contaminated. The solid sample is conservative for this tank. To ensure a conservative estimate, residual mass from Tank WM-182 was used for tanks that have not yet been decontaminated, including WM-190. The estimated mass of solid residuals by tank are as follows: Tank WM-180 (542 kg), Tank WM-181 (246 kg), Tank WM-182 (1,238 kg), Tank WM-183 (702 kg), Tank WM-184 (558 kg), Tank WM-185 (720 kg), and Tank WM-186 (334 kg).

- Analytical data are the primary source used to develop the radionuclide tank inventory. When these analytical data are not available, the ORIGEN2 model is used to predict the inventories.

- A ^{137}Cs ratio of 1.8 was established for solids. This value is the mean of all solid samples collected from Tank WM-188. These samples were collected in 1999, prior to tank cleaning. Use of this ratio increases the total radioactivity of all the radionuclides that were not detected.

- Although additional tank residuals may be removed during the grouting process (as discussed in Subsection 2.3.1), the residual waste inventory does not take credit for this additional waste removal.

- Actual analytical values from each tank were used for the ^{137}Cs ratio for liquids.

- Based on mass, WM-182 has the highest activity of any tank cleaned to date.

- An estimated density of 1.4 g/mL was applied for tank solids (EDF-TST-001, 2000).

Table 1 presents an example of the residual waste inventory at closure for a 300,000-gal tank. The inventory for Tank WM-182 is shown because this inventory contains the largest amount of residual radioactivity of the cleaned tanks. Inventories at closure for all tanks are presented in Appendix A. The analytical data indicate the decontamination efficiency is greater than planned, as shown in Subsection A-2 of Appendix A. The residual waste inventory at closure indicates that approximately 2,400 Ci remain in Tank WM-182. The total amount of radioactivity is associated almost entirely with residual solids. Approximately 2 Ci remain in the liquid in Tank WM-182.

Table 1. Residual waste inventory at closure for Tank WM-182.

Radionuclide	Residual Liquids (Ci)[a,b]	Residual Solids (Ci)[a,b]	Total Residuals (Ci)[c]
^{241}Am	5.30E–04	5.43E+00	5.43E+00
^{137m}Ba[d]	1.11E+00	1.14E+03	1.14E+03
^{242}Cm	6.58E–07	1.32E–03	1.32E–03
^{137}Cs	1.11E+00	1.14E+03	1.14E+03
^{14}C	5.39E–08	4.90E–06	4.96E–06
^{129}I	1.12E–06	7.73E–04	7.74E–04
^{3}H	1.66E–05	7.17E–01	7.17E–01
^{94}Nb	4.03E–05	2.06E–01	2.06E–01
^{59}Ni	1.25E–05	2.51E–02	2.51E–02
^{63}Ni	1.43E–03	2.86E+00	2.86E+00
^{237}Np	2.71E–07	4.70E–02	4.70E–02
^{238}Pu	2.47E–03	1.14E+01	1.14E+01
^{239}Pu	2.44E–04	3.40E+00	3.40E+00
^{240}Pu	6.74E–04	1.35E+00	1.35E+00
^{241}Pu	5.89E–04	1.95E+01	1.95E+01
^{242}Pu	4.93E–07	9.87E–04	9.88E–04
^{90}Sr	2.41E–01	2.32E+01	2.34E+01
^{99}Tc	4.54E–05	7.64E–01	7.64E–01
^{90}Y	2.41E–01	2.32E+01	2.34E+01
Total[e] Ci all radionuclides	3	2,391	2,394

a. Radionuclide inventories are based on (1) heel residuals that are estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points of known height, (2) best estimated radionuclide concentrations from past and recent samples as calculated in the associated engineering design file (Portage 2005c), and (3) radioactive decay to 2012.

b. Analytical results for Tank WM-182 were used to calculate the liquids inventory at closure for ^{241}Am, ^{14}C, ^{137}Cs, ^{154}Eu, ^{3}H, ^{129}I, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{125}Sb, ^{90}Sr, ^{99}Tc, ^{234}U, and ^{90}Y. Analytical results for Tank WM-183 were used to calculate the solids inventory at closure for ^{241}Am, ^{137m}Ba, ^{60}Co, ^{137}Cs, ^{129}I, ^{94}Nb, ^{238}Pu, ^{239}Pu, ^{125}Sb, ^{90}Sr, ^{99}Tc, ^{234}U, and ^{90}Y. Inventories at closure for all remaining nuclides were calculated using the methodology presented in Subsection 2.4.1.

c. Radionuclides decayed to 2012.

d. A 1:1 ratio is assumed for ^{137}Cs to ^{137m}Ba as a conservative estimate of radionuclide inventory; based on decay probability, ^{137m}Ba is approximately 95% of the ^{137}Cs inventory.

e. Radionuclides shown are contributors in PA dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides. This inventory is based on the 95% upper confidence limit of the radionuclide concentrations from liquid samples and an average of the estimate of remaining mass. Since only one sample of solid residual was able to be obtained, that concentration was used with the estimated remaining mass to determine the total solid residual.

Tank volume estimates in the 300,000-gal tanks for solid residual and interstitial water used in the residual waste inventory at closure are based on viewing videotapes of the tanks taken before, during, and after the final cleaning and sampling events. From the videotapes, residual solid waste depths were estimated by comparing the solids depths to the cooling coil support brackets and associated welds. Depth assumptions were based on the bottom weld and stainless steel bracket thickness measuring 0.97 cm (0.38 in.). Close-ups from the video were critical in determining the depths of waste next to the brackets and areas where no apparent solid residuals were observed. Depths of solid waste ranged from 0 to 0.97 cm (0 to 0.38 in.). Areas where no solids were present on the bottom of the tank were apparent by the reflection from the stainless steel bottom. The volume of solids and the density of the solid material were used to determine the mass of residual in each tank (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g). Videos and photographs of the tank walls show staining and discoloration, and no discernible buildup of waste residuals. Therefore, no source term for the tank walls was included in the tank inventory. Inventories at closure for all tanks are provided in Appendix A.

Residual solids depth measurements were recorded and entered into an AutoCAD 2004 drawing in association with the cooling coil support locations. Cartesian coordinates (x and y) were determined for each cooling coil support and associated residual depth measurement. The residual solids depth data and associated Cartesian coordinates for the cooling coil supports were then exported to Surfer 8. The residual solids volume was estimated using Kreiging methods in Surfer 8, with point-Kreiging being the preferred gridding method. A simple variogram model was used with linear interpretation using a slope equal to 1.0 and anisotropy equal to 1.[v]

Contour, post, and surface plot maps are provided for the residual solids volume estimates in the engineering design files prepared for each tank volume (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g). Residual solid volume estimates were determined using Surfer 8 with 0 cm (0 in.) as a lower surface boundary and the Trapezoid Rule[w] as the preferred volume estimate method. The depth of liquid above the interstitial waste was estimated to be approximately 3 cm (1 in.) by comparison to the height of the steam-jet lines in the post-decontamination video.

The residual waste inventory at closure assumes that the same degree of radionuclide removal will be achieved in the tanks remaining to be cleaned (WM-187, WM-188, WM-189, and WM-190) as was achieved in the tanks that have been cleaned. As described in Subsection 2.2, Tank WM-190 is an emergency spare tank and has never been used to store waste. However, this tank was contaminated with a small volume of first-cycle extraction process waste when the waste passed inadvertently through a transfer valve. As noted previously, Tank WM-182 contains the largest amount of residual radioactivity of the cleaned tanks.

For reasons as explained below, these assumptions are conservative. Solids and liquids from the cleaned tanks have been transferred to Tank WM-187 via existing piping. The decontamination fluids in Tank WM-187 were evaporated to provide space for additional decontamination fluids as additional tanks were decontaminated and emptied. This tank will contain all of the solids removed from the cleaned tanks and some of the liquid. Most of the liquid has been evaporated using the evaporator tank system. Although combining all of the solids in one tank may make cleaning this tank slightly more difficult, the solids in the tanks have been found to be made up of small particles that are removed easily using the tank cleaning system. Particle size distribution studies performed on samples collected from Tanks WM-182

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v. Kreiging is an interpolation technique that uses variograms as weighting functions. Variograms are measures of the continuity of spatial phenomena expressed as an average squared difference between measured quantities at different locations. Anisotropy is a state in which a physical characteristic varies when measured in different directions.

w. The Trapezoid Rule is a way to approximate definite integrals.

and WM-183 conclude the particles that comprise the solids are small and easily redispersed after settling (INEEL 2000b). The basic findings of the particle size study were substantiated during the course of tank cleaning. The solids were dispersed easily using high-pressure water sprays, and the particles remained suspended in the liquid for a significant period of time before they settled to the bottom. The characteristics of the solid particles are not known to have been altered by transferring to Tank WM-187.

2.4.3 Residual Waste Inventory at Closure for 30,000-gal Tanks

The residual waste inventory at closure for the 30,000-gal tanks is based on analytical data from post-cleaning sampling. Solid residual samples were not collected because an adequate volume of material was not present in the tanks (Portage 2005h). A film layer was observed on the lower half of all four tanks that appeared to be algae or another form of biological growth. This film layer was clearly not residual solids as found in the 300,000-gal tanks and is not likely to contain any significant radioactivity. However, to establish a conservative estimate, the film is assumed to be 5 mil (0.005 in.) thick and contains the concentrations of radionuclides found in Tank WM-183 solid samples (Portage 2005h). The inventories for each 30,000-gal tank vary from 36.2 to 36.7 Ci. Table 2 presents the inventory for Tank WM-106, which has the highest remaining Ci content of the 30,000-gal tanks.

Table 2. Residual waste inventory at closure for Tank WM-106 (Portage 2005h).

Radionuclide[a]	Residual Liquids (Ci)[b]	Residual Solids (Ci)[b]	Total Residual (Ci)[c]
^{241}Am	1.85E–06	6.36E–03	6.36E–03
^{137m}Ba	2.96E–02	1.72E+01	1.72E+01
^{242}Cm	1.75E–08	1.99E–05	1.99E–05
^{137}Cs	2.96E–02	1.72E+01	1.72E+01
^{14}C	5.28E–10	7.41E–08	7.46E–08
^{129}I	2.98E–08	1.17E–05	1.17E–05
^{3}H	3.90E–06	1.08E–02	1.08E–02
^{94}Nb	1.87E–05	3.10E–03	3.12E–03
^{59}Ni	3.34E–07	3.79E–04	3.80E–04
^{63}Ni	8.75E–06	4.32E–02	4.32E–02
^{237}Np	3.51E–08	7.10E–04	7.10E–04
^{238}Pu	1.65E–05	1.73E–01	1.73E–01
^{239}Pu	2.66E–06	5.14E–02	5.14E–02
^{240}Pu	1.80E–05	2.04E–02	2.04E–02
^{241}Pu	1.22E–05	2.95E–01	2.95E–01
^{242}Pu	1.31E–08	1.49E–05	1.49E–05
^{90}Sr	2.84E–01	3.50E–01	6.33E–01
^{99}Tc	8.32E–05	1.15E–02	1.16E–02
^{90}Y	2.63E–02	3.50E–01	3.76E–01
Total[d] Ci all radionuclides	0.6	36.1	36.7

a. Radionuclides based on results from the 2003 TFF PA (DOE-ID 2003b).
b. Analytical results from sampling of Tank WM-106 were used to calculate the liquids inventory at closure for ^{241}Am, ^{14}C, ^{244}Cm, ^{60}Co, ^{137}Cs, ^{154}Eu, ^{3}H, ^{129}I, ^{94}Nb, ^{63}Ni, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{90}Sr, ^{99}Tc, ^{234}U, and ^{235}U. Analytical results for Tank WM-183 were used to calculate the solids inventory at closure for ^{241}Am, ^{137m}Ba, ^{60}Co, ^{137}Cs, ^{129}I, ^{94}Nb, ^{238}Pu, ^{239}Pu, ^{125}Sb, ^{90}Sr, ^{99}Tc, ^{234}U, and ^{90}Y. Inventories at closure for all remaining nuclides were calculated using the methodology presented in Subsection 2.4.1.
c. Radioactive decay to 2012.
d. Radionuclides shown are contributors in PA dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

2.4.4 Residual Waste Inventory at Closure for Sandpads

Tanks WM-182 through WM-190 rest on a 15-cm (6-in.) layer of commercial-grade sand overlaying a concrete slab approximately 0.76 m (2.5 ft) thick. A 15- by 15-cm (6- by 6-in.) concrete curb encloses the sandpad area. The commercial-grade sand occupies approximately 23.39 m^3 per tank and provides a cushion for the stainless steel tanks to prevent shifting. A drain-sump system removes liquid (the liquid that accumulates is mostly rainwater; a small amount of liquid waste from secondary containments does enter vault sumps if a valve leaks or during some maintenance activities) that accumulates in the vaults. Typically, the vault area receives an accumulation of rainwater and snowmelt in the spring each year.

The sandpads at the TFF are between the tanks and vault floors. The sandpad at the bottom of the concrete vault was designed to cushion the tank bottom. Additionally, the curb has drain holes to allow liquid to drain from the sandpad beneath the tanks to the sumps outside the curb. The sumps were periodically used to remove liquid that had accumulated in the vaults. Liquid was present in the vault from sources such as precipitation and irrigation activities. The sandpads beneath Tanks WM-185 and WM-187 are contaminated due to accidental releases into the vaults in March 1962. These are the only tanks where such an incident occurred. A description of the back-siphoning event into the tanks is available in Latchum et al. (1962). Before and after these releases, water from precipitation, spring runoff, and irrigation was pumped out of the tank vaults at least semiannually. The residual inventory predicted for 2012 is based on 38 "flushing" events when water infiltrated to the vault from leaks in the tank/vault roof and was then jetted out of tank vaults. The amount of water collected in the vaults is documented in the INEL internal memorandum, "HLLW Tank Sump Transfer History."[x] The report documents over 100 transfers from tank vaults associated with WM-185 and WM-187. Reports are not available before 1962 but water transfers from the vaults are known to have occurred. Water transfers continue from vault sumps. Thirty-eight events correspond with each year from the back-siphoning events in 1962–2000. Water was jetted from the tank vault sumps at least twice yearly and will continue until each tank is closed. The actual number of water transfers from the tank vaults and associated leaching of radionuclides from the sandpad likely exceeds 130 for each vault to date.

The results from the analysis are provided for each radionuclide in the year 2012, which is the expected closure date. These activities have been estimated by modeling diffusion of radionuclides from the liquid into the sandpad for the contaminating event in 1962.

The approach uses diffusion of radionuclides to the sand. The sandpad was subsequently flushed by precipitation infiltration through the roof of the tank and radionuclides were decayed to 2012. The approach assumes radionuclides enter the sandpad through diffusion at the curb/sandpad interface. This assumption is reasonable since the contaminated liquid, which entered the vault in 1962, appears to have filled the concrete vault from the bottom toward the vault top (i.e., there was no horizontal pressure gradient driving the liquid through the sandpad). The radionuclides entered the sandpad by diffusion. Once contained in the sandpad, radioactive decay and periodic (annual) flushing are modeled to determine the present-day activities. Flushing is assumed to have occurred through 2000 (38 flushings) and then the inventory is decayed over 12 years to estimate 2012 activities of all radionuclides. The diffusion model is used because before the vault partially filled with waste, the vault was filled with water from a spring runoff event. The sandpad was saturated with water because water had filled a portion of the vault recently and in the past. The back-siphoning event that caused the waste to enter the vault was a result of jetting the water to the tank.

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x. Rebish, K. J., INEL, to J. M. Roberts and B. H. O'Brien, INEL, October 15, 1993, "HLLW Tank Sump History," KJR-10-93.

After the accident, the mass of each radionuclide in the sandpad is assumed to be impacted by radioactive decay and flushing. These processes are modeled by assuming the mass of radionuclides in the sandpad undergoes radioactive decay for a period of time representing the length of time between flushing events. Then the system is flushed. During flushing, the sandpad is assumed to be saturated and the radionuclides are partitioned at equilibrium between the liquid and solid phases according to their sorption coefficients (K_d values). The flushing is assumed to remove all liquid and radionuclides that have partitioned into the liquid phase from the sandpad except for the residual liquid. Thus, the radionuclides remaining in the sandpad are contained in the residual liquid and sorbed onto the sand. The mass of radionuclides in the sandpad is again calculated assuming radioactive decay occurs over a period of time that represents the time between flushing events. The cycle of modeling radioactive decay and flushing is repeated until 2000. In the absence of data, it was assumed that the flushing events occurred once a year for 38 years (i.e., once a year the seasonal effects of precipitation and irrigation are purged from the vaults). After activities in 2000 are computed, the results are decayed over 12 years to 2012, which corresponds to the time of TFF closure.

To determine the mass of radionuclides in the sandpad following the initial event in 1962, the radioactive decay and flushing of the sandpad is modeled. A FORTRAN computer code was developed to model radioactive decay and then flushing of the sandpad based on partition coefficients and the volume of sand, void space, and residual saturation. The mathematical formulation of this approach is provided below; the FORTRAN source code is presented in Appendix A of the PA (DOE-ID 2003b).

The accidental spill in 1962 is modeled assuming the liquid is present in the vault for 24 hours. The 24-hour time is consistent with the estimated time from the beginning of the spill to the time when the spilled liquid was removed by pumping the vault (Latchum et al. 1962). The area of the flux surface (i.e., the liquid/sandpad interface) is assumed to be an annular region having a radius equal to the tank radius and a thickness equal to the maximum sandpad height. That is, this area is approximately equal to the area of a rectangle with a length of $2\pi r$, where r is the radius of the tank (7.6 m [25 ft]), and a width of 15 cm (6 in.) (i.e., the maximum estimated height of the sandpad).

The initial amount of each radionuclide in the tanks at the time of the accidental spills was evaluated with limited sampling of Tanks WM-185 and WM-187 on February 14, 1962. Due to the limited number of radionuclides provided by this sampling analysis, an alternative method using information from a leak from Tank WM-181 was used to determine the inventory (EDF-CPP-058, 1997). Using a pseudo Al-clad fuel with an initial ^{235}U enrichment of 93% and a burnup of the processed fuel of 18%, EDF-CPP-058 evaluated the expected radionuclide content of the tank. The fuel from MTR Cycle No. 198 (Dykes 1963) was taken as typical for the fuel processed. The reactor contained 4,842 g of ^{235}U and had 684 MWd of operation over a 417-day period. For calculation purposes, inventories were normalized to the activity in a typical 200-g element. The ORIGEN2 data were corrected to the concentration of ^{137}Cs in Tank WM-185 one month before the incident. Data for Tanks WM-185 and WM-187 were shown in the record of the incident (Latchum et al. 1962). Tank WM-185 was used because of the slightly higher ^{137}Cs concentration of 1.71 Ci/L.

These assumptions and the data used in the analysis are considered reasonably conservative. That is, available data that represent the modeled system are used. When data are not available, conservative assumptions are made. For example, the surface available for diffusive transport in this analysis is assumed to remove all radionuclides that reach the surface instantaneously (i.e., the liquid/sandpad interface). Thus, mass transfer is greater under these conditions than if a concentration gradient is assumed to exist in the sandpad. With this and other assumptions and the available data, the analysis is expected to provide a reasonable and conservative estimate of the mass of each radionuclide contained in the sandpad. That is, the actual sandpad inventories are not expected to be larger than these estimates.

The amount of radionuclides that have been flushed from the sand has not been measured directly and cannot be determined from the analysis of the liquid in the vaults because other sources (such as vault sumps) contribute to vault contamination. However, because of the numerous flushing events that have occurred, any additional cleaning by flushing would not provide a significant benefit.

Table 3 presents a conservative residual waste inventory at closure for the sandpads based on the analysis of the two contaminated sandpads. This inventory represents the sandpad assumed under Tanks WM-185 and WM-187 in the PA. The residual waste inventory at closure was established using various numerical models to predict the initial radionuclide concentrations, the subsequent removal by the periodic flushing of the sandpads, and radioactive decay to estimate concentrations in 2012 (DOE-ID 2003b; Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g).

Table 3. Sandpad residual waste inventory at closure (Ci per sandpad).

Radionuclides	Sandpad Inventory (Ci)[a]
^{241}Am[b]	1.89E+00
^{137m}Ba	1.65E+03
^{242}Cm	1.38E–05
^{137}Cs[b]	2.53E–06
^{14}C[b]	3.90E–07
^{129}I[b]	1.08E–06
^{3}H	3.10E–22
^{94}Nb	2.29E–02
^{63}Ni	1.69E–10
^{237}Np	3.71E–04
^{238}Pu[b]	5.06E–06
^{239}Pu[b]	1.57E+00
^{240}Pu[b]	3.54E–01
^{241}Pu	2.28E+00
^{242}Pu	5.68E–05
^{90}Sr[b]	2.49E+02
^{99}Tc[b]	2.02E–12
^{90}Y	2.49E+02
Total[c] Ci all radionuclides	3,850

a. Radioactive decay to 2012.

b. Radionuclides based on results from the 2003 TFF PA (DOE-ID 2003b).

c. Radionuclides shown are contributors in PA dose calculations or regulated by concentration limits in 10 CFR 61.55. The total is based on the entire inventory of radionuclides, not just the key radionuclides presented in this table.

2.4.5 Residual Waste Inventory at Closure for Contaminated Piping, Encasements, and Valve Boxes

Liquid waste transfers to, from, and between the tanks are managed through a series of lines, valves, and diversion boxes. Piping that penetrates tanks, vaults, or valve boxes is part of this TFF draft 3116 Determination. For each phase of the TFF closure, contaminated piping, encasements,[y] and valve boxes will be isolated and decontaminated. In general, pipelines entering or leaving the INTEC TFF will be isolated at the first valve box or manhole inside the TFF fence. Lines that begin and end within the TFF will be closed entirely; however, they may be closed in stages, depending on the TFF waste handling needs. Lines that enter the TFF boundary from outside will be closed at a convenient junction, such as a valve box. The remaining line section will be closed with the tank or facility from which it originates. Waste transfer piping will be decontaminated by rinsing each line with three volumes of water.

Valve boxes are underground, stainless steel-lined concrete boxes that house waste transfer valves through which liquid waste is routed. During each TFF closure phase, the inside surfaces of each valve box will be washed, and the valve box will be decontaminated with demineralized water. Based on past maintenance decontamination practices, this protocol will be adequate to clean the interior pipe surfaces that have contacted process solutions. Piping that penetrates tanks and encasements that penetrate vaults will be filled with grout to the extent practical (INEEL 2000c).

Valve boxes and piping encasements that penetrate tanks or vaults will be drained of liquids resulting from decontamination efforts or leakage (DOE-ID 2003a) and then grouted along with their drain lines. A residual waste inventory at closure is not calculated for valve boxes because they do not normally contain process solutions; they are designed so that any leakage into the valve box would drain to sumps, which is then jetted to tanks; and the surface area of the valve boxes is small compared with the piping and tanks. Valve boxes have been (or will be) decontaminated and samples collected of the remaining water in the sump, but an inventory was not prepared. The valve box sumps are approximately 0.3 m (1 ft) square and have an outlet that is approximately 5–8 cm (2–3 in.) above the bottom of the sump. The liquid that is sampled is taken from the 5–8 cm (2–3 in.) of volume in the sump bottom. Almost all of the liquid is removed by sampling.

The TFF process lines are stainless steel pipe with secondary containment or encasements (see Subsection 2.1). Process waste lines and associated encasements are covered with 1.5–4.6 m (5–15 ft) of soil with about 30% being within 3 m (10 ft) of the surface (INEEL 2000c). During TFF closure, the active encasements will be decontaminated using a pressure-washing system. Most pipe encasements open into the associated valve boxes or vaults or have drains that drain into the associated vault. Each encasement is sloped to allow drainage and is flushed during tank isolation activities. The grout will be pumped into an encasement and will run or be pushed to the end of the encasement (INEEL 2000c). A residual waste inventory at closure is not calculated for pipe encasements. The pipe encasements are similar to valve boxes in that they do not normally contain process solutions; they are designed so that any leakage into the pipe encasements would drain to sumps, which is then jetted to tanks; and the contamination in the secondary containment is addressed by using the safety factor of 500 established for the piping inventory. The safety factor of 500 is described in Subsection 2.3.1.

Tank vessel off-gas piping will not be flushed because the piping did not transport process solutions (INEEL 2000c). The vessel off-gas lines will be grouted and permanently capped. A residual

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y. Piping encasements are enclosures (secondary containment) surrounding the waste transfer piping to prevent escape of radioactive waste. Any fluid that leaked from a process line drained into an encasement and then into a valve box or vault sump.

waste inventory at closure is not calculated for tank vessel off-gas piping because the piping has never contained process solutions.

The residual waste inventory at closure for TFF piping is based on analytical results from sampling pipe sections that had been removed from the Tank WM-182 process waste lines after decontamination. The samples were obtained and analyzed as described in the associated SAP (INEEL 2001). From these results, the amount of residuals remaining in 3,231 linear m (10,600 linear ft) of process waste piping is calculated to be 15.5 kg of SBW solid residuals in the piping. The radionuclide concentration from the tank residual inventory is then apportioned to the mass to estimate a conservative residual waste inventory at closure for the piping. Table 4 shows the total radioactivity by radionuclide for the piping.

Table 4. Residual waste inventory at closure for piping.

Radionuclides	Piping Inventory (Ci)[a,b]
^{241}Am[c]	5.28E–03
^{137m}Ba	1.43E+01
^{242}Cm	1.65E–05
^{137}Cs[c]	1.43E+01
^{14}C[c]	6.21E–08
^{129}I[c]	9.70E–06
^{3}H	1.08E–02
^{94}Nb	2.58E–03
^{59}Ni	3.15E–04
^{63}Ni	3.58E–02
^{237}Np	5.89E–04
^{238}Pu[c]	1.43E–01
^{239}Pu[c]	4.27E–02
^{240}Pu[c]	1.69E–02
^{241}Pu	2.44E–01
^{242}Pu	1.24E–05
^{90}Sr[c]	2.93E–01
^{99}Tc[c]	9.57E–03
^{90}Y	2.93E–01
Total[d] Ci all radionuclides	30

a. Radionuclide inventories are based on (1) residual solids in piping at 0.305 g/m and 3,231 linear m (10,600 linear ft) of piping, (2) WM-182 radionuclide concentrations in the SBW solids, and (3) radioactive decay to 2012.

b. Analytical results for Tank WM-183 were used to calculate the solids inventory at closure for ^{241}Am, ^{137m}Ba, ^{60}Co, ^{137}Cs, ^{129}I, ^{94}Nb, ^{238}Pu, ^{239}Pu, ^{125}Sb, ^{90}Sr, ^{99}Tc, ^{234}U, and ^{90}Y. Inventories at closure for all remaining nuclides were calculated using the methodology presented in Subsection 2.4.1.

c. Radionuclides based on the 2003 TFF PA (DOE-ID 2003b).

d. Radionuclides shown are contributors in PA dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

3. SECTION 3116 OF THE RONALD W. REAGAN NATIONAL DEFENSE AUTHORIZATION ACT FOR FISCAL YEAR 2005

Section 3116(a) of the NDAA provides:

> In General—Notwithstanding the provisions of the Nuclear Waste Policy Act of 1982, the requirements of section 202 of the Energy Reorganization Act of 1974, and other laws that define classes of radioactive waste, with respect to material stored at a Department of Energy site at which activities are regulated by a covered State pursuant to approved closure plans or permits issued by the State, the term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy (in this section referred to as the "Secretary"), in consultation with the Nuclear Regulatory Commission (in this section referred to as the "Commission"), determines—
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical; and
>
> (3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, and will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations, and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and
>
> (iii) pursuant to plans developed by the Secretary in consultation with the Commission.

4. WASTE DOES NOT REQUIRE PERMANENT ISOLATION IN A DEEP GEOLOGIC REPOSITORY FOR SPENT FUEL OR HIGH-LEVEL RADIOACTIVE WASTE

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines—
>
> > (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste.

Section 3116(a) clarifies the Secretary's authority, in consultation with the NRC, to determine that certain waste from reprocessing that meets the criteria set out in the section should not be classified as "high-level radioactive waste." Section 3116(a) sets out two specific criteria for making this determination in Clauses (2) and (3). Clause (2) requires the DOE to remove highly radioactive radionuclides to the maximum extent practical. Clause (3) generally mirrors the criteria that the NRC has established for determining whether waste qualifies for land disposal as LLW (10 CFR 61.55). This clause provides that disposal of the waste must meet the NRC performance objectives of 10 CFR 61, Subpart C, and that the waste must not exceed the concentration levels for Class C waste in 10 CFR 61.55 or the Secretary must consult with NRC concerning DOE's disposal plans.

Clause (1), noted above, is a broader criterion for the Secretary, in consultation with the NRC, to consider whether, notwithstanding that waste from reprocessing meets the other two criteria, there are other considerations that, in the Secretary's judgment, require its disposal in a deep geologic repository. Generally, such considerations would be an unusual case because waste that meets the third criterion would be waste that will be disposed of in a manner that meets the 10 CFR 61, Subpart C performance objectives and either falls within one of the classes set out in 10 CFR 61.55 that the NRC has specified are considered "generally acceptable for near-surface disposal" or for which the Secretary has consulted with NRC concerning DOE's disposal plans. As the NRC explained in *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services)* (CLI-05-05, 2005), the 10 CFR Part 61, Subpart C performance objectives in turn "set forth the ultimate standards and radiation limits for (1) protection of the general population from releases of radioactivity; (2) protection of individuals from inadvertent intrusion; (3) protection of individuals during operations; and (4) stability of the disposal site after closure." It follows that if disposal of a waste stream in a facility that is not a deep geologic repository will meet these objectives, in the ordinary case that waste stream does not "require disposal in a deep geologic repository" because non-repository disposal will be protective of public health and safety.

It is possible that in rare circumstances a waste stream that meets the third criterion might have some other unique radiological characteristic or may raise unique policy considerations that warrant its disposal in a deep geologic repository. Clause (1) is an acknowledgement by Congress of that possibility. For example, the waste stream could contain material that, while not presenting a health and safety danger if disposed of at near- or intermediate-surface, nevertheless presents non-proliferation risks that the Secretary concludes cannot be adequately guarded against absent deep geologic disposal. Clause (1) gives the Secretary, in consultation with NRC, the authority to consider such factors in determining whether waste that meets the other two criteria needs disposal in a deep geologic repository in light of such considerations.

That is not the case here. As will be demonstrated later in this draft 3116 Determination, disposal of the TFF residual waste and the TFF tank system at closure will meet the performance objectives of 10 CFR 61, Subpart C so as to provide for the protection of the public health and the environment. Accordingly, the waste does not require disposal in a deep geologic repository due to risk to public health and safety. Furthermore, disposal of the TFF residual waste and TFF tank system does not raise any unique considerations that, notwithstanding these demonstrations, require its permanent isolation in a deep geologic repository. Accordingly, the TFF residual waste and TFF tank system qualify for classification as other than HLW under Clause (1) of Section 3116(a).

5. WASTE HAS HAD HIGHLY RADIOACTIVE RADIONUCLIDES REMOVED TO THE MAXIMUM EXTENT PRACTICAL

Key Points

- The list of highly radioactive radionuclides for the TFF residual waste and TFF tank system describes the radionuclides that could reasonably be expected to exist in the waste and that, using a risk-informed approach, contribute significantly to the radiological risk to workers, the public, and the environment, taking into account scientific and health physics principles, knowledge, and expertise.
- In demonstrating that radionuclides will be removed to the maximum extent practical as required by Section 3116, the same degree of radionuclide removal is expected for the four 300,000-gal tanks remaining to be cleaned as that achieved in the seven tanks that have been cleaned, as described in Subsection 5.2.
- The demonstration that radionuclides will be removed to the maximum extent practical does not include the amount of additional residual waste that may be removed from the cleaned 300,000-gal tanks during grouting operations.

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical.

5.1 Highly Radioactive Radionuclides

5.1.1 Approach

Based on consultations with NRC, DOE views "highly radioactive radionuclides" to be those radionuclides that, using a risk-informed approach, contribute most significantly to radiological risk to workers, the public, and the environment. Table 5 lists the radionuclides for the grouted (stabilized) TFF residual waste and TFF tank system at closure and disposal that DOE has determined, on the basis of a risk-informed approach, contribute significantly to radiological risk to workers, the public, and the environment, taking into account scientific and health physics principles, knowledge, and expertise.

The list of highly radioactive radionuclides was developed, beginning with the entire inventory of radionuclides stored at INTEC (ICP 2005c).[z] The DOE reviewed this inventory of radionuclides and

z. The DOE has reviewed the inventory of 145 radionuclides in the TFF, as reflected in the current inventory for INTEC.

identified those radionuclides in Tables 1 and 2 of 10 CFR 61.55,[aa] as well as any additional radionuclides that may be important to meeting the performance objectives in 10 CFR 61, Subpart C because they contribute to the dose to workers, the public, and/or the inadvertent intruder (for one or more reasonable intruder scenarios) in the expected and degraded cases, using sensitivity analyses. In DOE's view, this approach results in a risk-informed list of highly radioactive radionuclides that includes: those short-lived radionuclides that may present risk because they produce radiation emissions that, without shielding or controls, may harm humans simply by proximity to humans without inhalation or ingestion; and those long-lived radionuclides that persist well into the future, may be mobile in the environment, or may pose a risk to humans if inhaled or ingested. This list includes the short-lived fission products ^{137}Cs and ^{90}Sr and their equilibrium daughters, ^{137m}Ba and ^{90}Y, that are responsible for the bulk of activity in the TFF tank system inventory. Based on process knowledge, ^{137}Cs and ^{90}Sr and their equilibrium daughters, ^{137m}Ba and ^{90}Y, account for approximately 99% of the activity in the TFF tank waste (ICP 2005c). This list also includes the long-lived actinides (^{241}Am, ^{242}Cm, ^{237}Np, and the plutonium isotopes) that are responsible for approximately 0.08% of the activity. The other radionuclides included in the list are ^{14}C, ^{3}H, ^{129}I, ^{94}Nb, ^{59}Ni, ^{63}Ni, and ^{99}Tc, which contribute approximately 0.05% of the activity. The list of highly radioactive radionuclides (shown in Table 5) account for approximately 99% (see Appendix A) of the radioactivity in the TFF tank waste as of April 2005, based on process and sampling knowledge.

5.1.2 Radionuclides from Performance Assessment

As explained above, DOE has included in the list of highly radioactive radionuclides (shown in Table 5) that may be important to meeting the performance objectives of 10 CFR 61, Subpart C because they contribute to the dose to workers, the public, and/or the inadvertent intruder based on the TFF PA, which includes a sensitivity and uncertainty analyses (DOE-ID 2003b). The DOE used the following approach in the PA to focus on those radionuclides that contribute to the dose for various pathways. The results of the PA show that ^{137m}Ba and ^{90}Y should also be included in the list of highly radioactive radionuclides shown in Table 5.

5.1.2.1 Groundwater Radionuclide Screening. Screening of radionuclides and numerical modeling and dose assessment was performed to form the list of highly radioactive radionuclides. The TFF inventory includes several radionuclides that are significant in terms of projected receptor doses from the groundwater pathway. Therefore, screening analysis methods were investigated for use in the TFF PA to focus the groundwater analysis on highly radioactive radionuclides. The groundwater screening analysis considered that active institutional control will be maintained over the disposal site for 100 years after facility closure and, furthermore, that the performance of the facility will be monitored to detect and/or prevent significant releases of radionuclides to the environment throughout the period of active institutional control. This consideration eliminates from concern any radionuclide with a half-life of less than 5 years because the inventory in the waste at 100 years after facility closure will be reduced to innocuous levels by radioactive decay. However, radionuclides with half-lives of less than 5 years cannot be neglected if the radionuclide appears in a decay chain. This is because its activity may increase with time due to decay of a parent radionuclide, unless the parent is also short-lived.

aa. Although Tables 1 and 2 in 10 CFR 61.55 specify concentration limits for certain radionuclides in the form of activated metal, DOE has included such radionuclides, if present in the waste, in the list of "highly radioactive radionuclides" as it exists in the waste, without regard to whether such radionuclides are in the form of activated metal. Consistent with Table 1, DOE has excluded alpha-emitting transuranic nuclides with half-lives of 5 years or less from the list of highly radioactive radionuclides. Some of the radionuclides listed as highly radioactive radionuclides in this draft 3116 Determination may not be listed in other 3116 Determinations if such radionuclides are not present in the waste or do not contribute to dose to the worker, the public, or the intruder.

Table 5. List of highly radioactive radionuclides.

Radionuclide	Radionuclide Half-Life (yr)	Long-Term Radiation Hazards	Short-Term Radiation Hazards
^{241}Am[a,b]	4.3E+02	X	
^{14}C[a,b]	5.7E+03	X	
^{242}Cm[b,c]	4.5E–01	X	
^{60}Co[d]	5.3E+00		X
^{137}Cs[a,d]	3.0E+01		X
^{137m}Ba[a]	4.9E–06		X
^{3}H[a,d]	1.2E+01		X
^{129}I[a,b]	1.6E+07	X	
^{94}Nb[b]	2.0E+04	X	
^{59}Ni[b]	7.5E+04	X	
^{63}Ni[b]	1.0E+02		X
^{237}Np[a,b]	2.1E+06	X	
^{238}Pu[a,b]	8.8E+01	X	
^{239}Pu[a,b]	2.4E+04	X	
^{240}Pu[a,b]	6.6E+03	X	
^{241}Pu[b,c]	1.4E+01	X	
^{242}Pu[b]	3.8E+05	X	
^{90}Sr[a,d]	2.9E+01		X
^{90}Y[a]	7.3E–03		X
^{99}Tc[a,b]	2.1E+05	X	

a. Highly radioactive radionuclides based on dose assessment results from the 2003 PA (DOE-ID 2003b).

b. Taken from Table 1 of 10 CFR 61.55.

c. Daughter product of long-lived radionuclide.

d. Taken from Table 2 of 10 CFR 61.55.

The use of the 5-year half-life screening criteria results in the elimination of the following radionuclides from further consideration: ^{102}Rh, ^{119m}Sn, ^{134}Cs, ^{150}Eu, ^{153}Gd, ^{155}Eu, ^{55}Fe, and ^{171}Tm. In addition, the following radionuclides in short decay chains were eliminated from further analysis since the parent and progeny each have half-lives of less than 5 years: $^{106}Ru/^{106}Rh$, $^{109}Cd/^{109m}Ag$, $^{110m}Ag/^{110}Ag$, $^{125}Sb/^{125m}Te$, and $^{144}Ce/^{144m}Pr/^{144}Pr$. Additional radionuclides were screened from further consideration since their half-lives indicate that they are either stable or have such long half-lives that their specific activity, and thus contribution, to dose would be insignificant. These radionuclides include ^{142}Ce (5.0E+16 yr), ^{149}Sm (1.0E+15 yr), ^{144}Nd (2.4E+15 yr), and ^{148}Sm (2.0E+14 yr).

The screening procedure used for the TFF inventory was based first on the concentrations of radionuclides in the waste pore water that would give an annual effective dose equivalent of 4 mrem/yr from consumption of 70 oz/d (i.e., 200 gal/yr) of contaminated water. The 4-mrem/yr standard was used for screening each nuclide because this portion of the screening process does not consider transport from the waste form to the groundwater. A total of 29 radionuclides were retained after screening using the pore water concentrations.

The next step in the screening process involved evaluation of the release of radionuclides from the waste form and the resulting groundwater concentrations. Releases and groundwater concentrations were previously analyzed by numerical modeling. The results are presented in Appendix F of the PA (DOE-ID 2003b). Based on these analyses, only ^{129}I, ^{99}Tc, and ^{90}Sr were found to result in concentrations in groundwater that contribute significantly to doses (at least 99% of the all-pathways dose is attributable to these radionuclides).

The *Idaho High-Level Waste & Facilities Disposition Final Environmental Impact Statement* (DOE 2002) also conducted screening and groundwater transport analyses for the TFF. Their screening and analyses found that only ^{99}Tc and ^{129}I resulted in significant doses from the groundwater pathway. Therefore, based on previous modeling of the TFF presented in Appendix F of the PA, radionuclides (i.e., ^{129}I, ^{99}Tc and ^{90}Sr) were determined to result in appreciable groundwater concentrations and doses. Therefore, the groundwater analysis was focused on the three radionuclides and they were included in the list of highly radioactive radionuclides.

5.1.2.2 Intruder Pathway Radionuclide Screening. The intruder analysis considers that active institutional control will be maintained over the disposal site for at least 100 years after facility closure. This consideration eliminates from concern any radionuclide with a half-life of less than 5 years.

Using the 5-year half-life screening criteria results in the elimination of the following radionuclides from further consideration: ^{102}Rh, ^{119m}Sn, ^{134}Cs, ^{150}Eu, ^{153}Gd, ^{155}Eu, ^{55}Fe, and ^{171}Tm. In addition, the following radionuclides in short decay chains were eliminated from further analysis since the parent and progeny each have half-lives of less than 5 years: ^{106}Ru/^{106}Rh, ^{109}Cd/^{109m}Ag, ^{110m}Ag/^{110}Ag, ^{125}Sb/^{125m}Te, and ^{144}Ce/^{144m}Pr/^{144}Pr. Additional radionuclides were screened from further consideration because their half-lives indicate that they are either stable or have such long half-lives that their specific activity and thus, contribution to dose, would be insignificant. These radionuclides include ^{142}Ce (5.0E+16 yr), ^{149}Sm (1.0E+15 yr), ^{144}Nd (2.4E+15 yr), and ^{148}Sm (2.0E+14 yr).

Dose assessment was then used to define the radionuclides that would be included in the intruder scenarios. The inclusion of radionuclides is explained in Section 5 of the PA (DOE-ID 2003b). A review of the PA shows the dose contribution by highly radioactive radionuclides was a factor of 10 greater than the dose from the radionuclides excluded from further intruder scenario analysis. The air pathway analysis included ^{3}H and ^{14}C because these radionuclides are volatile or gaseous.

5.2 Radionuclide Removal to the Maximum Extent Practical

Section 3116(a) of the NDAA provides that certain waste resulting from reprocessing is not HLW if the Secretary, in consultation with the NRC, determines, among other things, that the waste has had highly radioactive radionuclides removed "to the maximum extent practical." This subsection discusses the basis on which the Secretary may conclude that DOE's approach for removing the highly radioactive radionuclides listed above meets this criterion.

Removal to the maximum extent "practical" is not removal to the extent "practical" or theoretically "possible." Rather, a "practical" approach to removal is one that is "adapted to actual conditions" (Fowler 1930); "adapted or designed for actual use" (Random House 1997); "useful" (Random House 1997); selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" (Random House 1997); fitted to "the needs of a particular situation in a helpful way" (Cambridge 2004); "effective or suitable" (Cambridge 2004). Therefore, the determination as to whether a particular highly radioactive radionuclide will be removed to the maximum extent practical will vary

from situation to situation, not only on the available technologies but also on the overall costs and benefits[bb] of deploying a technology with respect to a particular waste stream. The "maximum extent practical" standard contemplates room for exercising expert judgment in weighing several factors. Such factors may include environmental, health, timing, or other exigencies; the risks and benefits to public health, safety, and the environment arising from further radionuclide removal as compared with countervailing considerations that may ensue from not removing or delaying removal; the reasonable availability of proven technologies; the usefulness of such technologies; and the sensibleness of using such technologies. What may be removal to the maximum extent practical in a particular situation or at one point in time may not be that which, on balance, is practical, feasible, or sensible in another situation or at a prior or later point in time.

Although not an issue here, DOE notes that that it may not be practical to undertake any further removal of certain radionuclides because further removal is not sensible or useful in light of the overall benefit to human health or the environment. As a general matter, such a situation may arise if certain radionuclides are present in such extremely low quantities that they make an insignificant contribution to potential doses to workers, the public, and the hypothetical human intruder.[cc]

Radionuclides have been removed from the TFF tank system to the maximum extent practical by removing liquid waste from the TFF tanks for calcining operations (described in Subsection 2.2) and by SBW removal and tank cleaning activities. The same degree of radionuclide removal is expected for the four 300,000-gal tanks and ancillary equipment remaining to be cleaned.[dd] To demonstrate compliance with Section 3116(a)(2), the list of highly radioactive radionuclides identified in Subsection 5.1, the total inventory of radionuclides in the TFF waste stream, and the residual waste inventory at closure and disposal are used to determine the effectiveness of radionuclide removal. These radionuclides in the list of highly radioactive radionuclides in Table 5 are listed or referred to in 10 CFR 61.55, "Waste Classification," are significant contributors to the PA dose calculations and contribute to the radionuclide inventory (DOE-ID 2003c). Further removal of the residual waste inventory is not practical, as discussed below.

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bb. While prior NRC and DOE requirements for waste determinations called for removal "to the maximum extent *technically* and *economically* practical" (Bernero 1993; DOE M 435.1-1, 2001), Section 3116 omits these adverbs, thereby suggesting that a broad range of considerations, including but not limited to technical and economic practicalities, may appropriately be taken into account in determining the extent of removal that is practical.

cc. The DOE normally would view radionuclides as making an insignificant contribution if the contribution to dose from those radionuclides, in both the expected case and fully degraded case using sensitivity analysis, does not exceed any of the following: (1) 10% of the 25-mrem/yr all-pathways annual dose to the public, (2) 10% of the DOE 100-mrem annual dose limit to the intruder (under all reasonable intruder scenarios), (3) 10% of the DOE 500-mrem acute dose limit to the intruder (under all intruder scenarios), and (4) 10% of the annual worker dose in the relevant provisions of 10 CFR 20. For perspective, DOE would also consider the extent to which such radionuclides contribute to an annual dose of 4 mrem to the receptor from the groundwater pathway. This methodology is based on NRC consultation and is intended to be consistent with the guidance and general approach in Volume 2 of NUREG-1757, *Consolidated NMSS Decommissioning Guidance* (NRC 2003), which explains that "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10% of the dose criteria to be insignificant contributors." The above-reference NUREG, which applies to NRC licensees, is being used only as general guidance, and DOE's use of this NUREG as guidance should not be construed to suggest that it is a requirement under Section 3116 of the NDAA or that either the NUREG or 10 CFR 20, Subpart E is applicable in the 3116 context.

dd. The same degree of removal is expected because the waste has not been and will not be radically altered. The particle size and density of the solids are not affected by tank-to-tank transfers during cleaning. Moreover, transfers between tanks have been a common practice in the past.

The TFF tank system cleaning activities result in removal of radionuclides to the maximum extent practical. During tank cleaning activities for the 300,000-gal tanks, waste is removed from each tank in two phases, as described in Subsection 2.3.1. First, waste is removed using existing waste transfer equipment. Then, additional waste is removed using the tank cleaning system and modified waste transfer equipment. This second phase of waste removal continues until radiation levels near the steam-jet transport line decrease to the lowest constant value, indicating that the cleaning and waste transfer operations are no longer effective. Any remaining tank waste is regarded as waste residual. The inventory in the piping is insignificant (30 Ci) when compared to that of other system components (DOE-ID 2003b). The sandpad inventory is conservative, and no practical means exist to remove additional Ci from the sandpads that are located under the 300,000-gal tanks. A review of the tank construction and design, which includes access risers and the dimensions of the tank and sandpad, do not allow for removal of sand. The amount of radionuclides that have been flushed from the sand has not been measured directly and cannot be determined from the analysis of the liquid in the vaults because other sources (such as vault sumps) contribute to vault contamination. However, because of the numerous flushing events that have occurred, any additional cleaning by flushing would not provide a significant benefit. Therefore, tank cleaning is the only practical source for inventory reduction. Approximately 57% of the total system residual waste inventory at closure is expected to be contained in the fine layer of solids that is distributed unevenly over the bottom of the 300,000-gal tanks (see photographs in Figures 11 and 12) (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g). The remaining 43% is attributed to the conservative estimated inventory (as discussed below) for the contaminated sandpads (WM-185 and WM-187).

The tank cleaning process is thorough, and the process is reviewed during cleaning to maximize effectiveness. Videotapes are reviewed by observing benchmarks in the bottom of the tanks. The support brackets for cooling coils (0.97 cm [0.38 in.]) or the welds in the bottom (0.32 cm [0.13 in.]) of the tank are compared to the depth (if any) of solid residual. Operators and supervisors worked together to review videos of cleaning and to view real-time cleaning on cameras to maximize performance of the operators during the cleaning cycle. Cleaning activities are monitored using cameras and video recording. In addition, radiation levels are monitored on the steam-jet transport line to ascertain cleaning effectiveness. Monitoring the radiation levels near the transport line provides the cleaning system operators an indication of when tank cleaning should cease. As cleaning operations progress, the level of radiation observed near the transport line decreases asymptotically over time to a minimum radiation level that cannot be significantly reduced further with continued cleaning operations. When radiation levels decrease to this lowest constant value, cleaning is stopped, and the tanks are visually examined using remote video inspection. Use of the radiation monitor allows tank cleaning to proceed without repeated visual inspection or sample collection and aids in ensuring that as much waste as practical is removed from the tanks (Kimmitt 2002). Personnel review videotape of cleaning activities and compare the visual results to the radiation monitoring results to optimize solid residual removal in each tank. Based on observations made during tank cleaning activities performed to date, tank agitation by the directional nozzles is most effective when directional nozzles are used with approximately 5,000–7,000 gal of water in the tank for a short time (100–140 minutes). This technique ensures that solid particles are suspended in the liquid for a period comparable to the time required to remove the liquid and suspended solids by pumping (EDF-TST-001, 2000).

The optimization techniques developed during tank cleaning increases radionuclide removal efficiency. Monitoring the process waste lines is proving to be an excellent tool to determine when cleaning techniques are reaching the maximum effectiveness as practical. To illustrate, Figure 13 shows the radioactivity levels during Tank WM-182 cleaning operations. The graph shows that significant radioactivity was detected in the outlet process piping when cleaning first began. Peaks in radioactivity removed indicate when the steam jets were removing residuals. The troughs indicate when the steam jets were not in use and washball or directional nozzles were being used to clean tank internal surfaces or

suspend solids by agitation. Cleaning and pumping were often performed in unison. When the steam jet lost suction because liquid had been removed to a negligible level, pumping stopped until flush water from cleaning accumulated to a level that suction could resume. Near the end of the cleaning (after approximately 20,000 gal of flush water through directional nozzles), the radioactivity detected remained at or near the baseline level while cleaning and waste removal progressed. This condition was a clear indication that waste removal had been maximized. Subsequent video inspection and sampling and analysis confirmed the tank had been cleaned to the maximum extent practical. There is no evidence of any buildup of residual on the side walls after cleaning. Large areas of the tank floor are bare and some areas of the tank floor have a fine layer (approximately 0.97 cm [0.38 in.]). Additional spray cleaning was not able to remove this small quantity of residual.

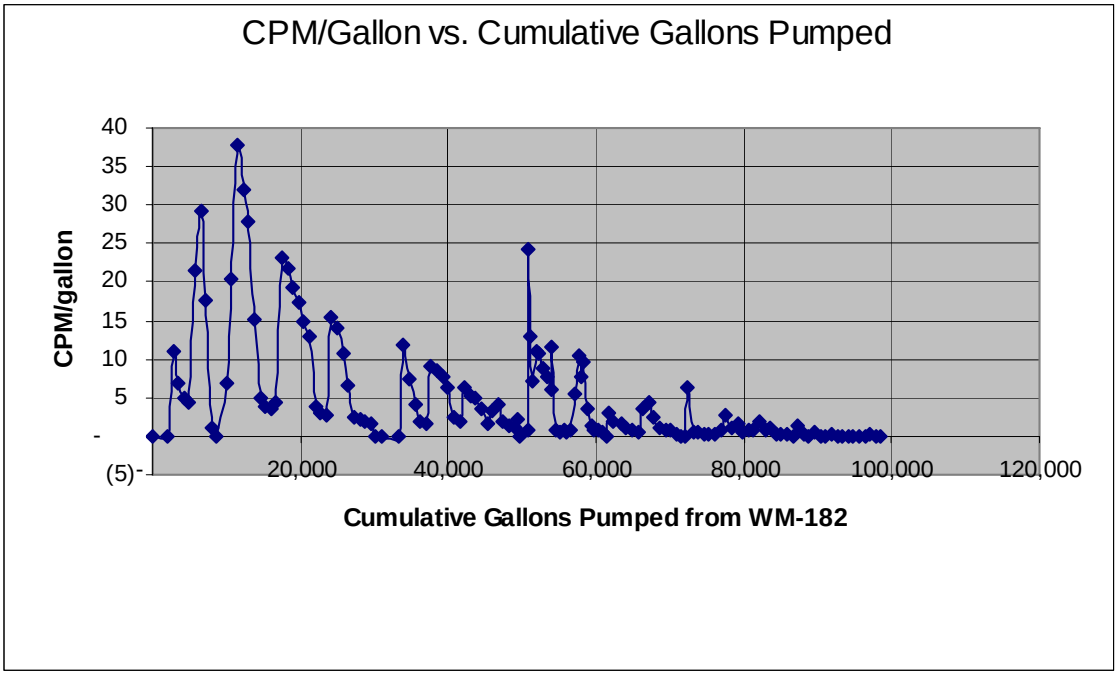

Figure 13. Radioactivity (counts per minute per gallon) measured during cleaning operations.

As discussed in the following subsections, further removal of this residual waste is not practical and any efforts to remove more of the small quantity of waste remaining would not significantly reduce the potential risk to the public. The INL Site and DOE have successfully evaluated, selected, and deployed a tank cleaning system so that radionuclides are removed to the maximum extent practical. Various cleaning technologies were considered for removing residuals effectively from the TFF with the goal of protecting public and occupational health and safety at the tank closure site. Technologies were evaluated by comparing information on effectiveness, maturity, and cost balanced against minimizing occupational radiation dose. As discussed in Section 2, the TFF tank cleaning technology was selected in coordination with the DOE TFA. The DOE commissioned this team to coordinate technology needs for DOE complex-wide radioactive tank waste remediation problems. The team considered both chemical and mechanical processes for cleaning tanks. The INL Site determined that chemical processes were not practical because the TFF waste is acidic and adding other chemicals would be unlikely to increase waste removal efficiency. Some mechanical cleaning systems were found to be applicable, and high-pressure water systems (which slurry the solids) and mechanical arms would perform well in the TFF tanks. The washball and directional spray nozzle high-pressure water systems were determined practical for

removing the small quantity of solids from the tank walls and for slurrying the solids on the tank bottoms. (See Appendix B for a discussion of the tank cleaning technology selection.)

5.2.1 Percentage of Radionuclides Removed

To quantitatively address how effective the waste removal techniques have been, a mass balance approach has been developed. Historically, radionuclides were removed from the total TFF waste stream by removing the liquid waste for the calcination process. The total inventory of radionuclides in the TFF waste stream has been identified and is provided in *Generation, Disposition, and Current Inventory of Radionuclides in the INTEC Tank Farm* (ICP 2005c). This inventory was identified by preparing a mass balance of radionuclides received from SNF reprocessing and sent to the TFF for storage and disposition. The mass balance relied on analytical data, numerical modeling, and data extrapolation to arrive at the total radioactivity present at INTEC from SNF reprocessing. The mass balance, which also considered nuclear material shipped to other DOE facilities for further processing, was used to determine the total radioactivity of waste generated from spent fuel reprocessing, which was 36 million Ci.

Table 6 shows the initial inventory of all waste at INTEC (decayed to 2012) from reprocessing by radionuclide; the estimated residual waste in the 300,000-gal tanks, 30,000-gal tanks, the two contaminated sandpads, and piping after cleaning at closure; and the percentage of radioactivity removed by treatment (calcination and tank cleaning), shipment of nuclear material, and radioactive decay to the date of tank closure (2012). This percentage is determined by comparing the initial INTEC inventory of the radioactivity (Ci) that has been sent to the TFF with the estimated residual waste inventory at closure. As discussed in Section 2, the residual waste inventory at closure relies on analytical data from samples obtained during recent tank cleaning operations. Cleaning in the remaining 300,000-gal tanks is assumed to be as effective as in those that have been cleaned already for the reasons explained earlier in this draft 3116 Determination. The physical properties (INEEL 2000b) of the solid residual are well known. Solids were effectively cleaned from the tanks. There is not any information that indicates the potential for the tank cleaning system to be ineffective in the remaining tanks (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g, 2005h). More volume of water or a longer cleaning time may be necessary because of the amount of solids now stored in remaining tanks (with the exception of Tank WM-190). The residual inventory for the four tanks that have not been cleaned is represented by the inventory in Tank WM-182 since it contains the largest Ci quantity of residual. Individual estimated tank inventories at closure are presented in Appendix A. Removal of residuals during tank cleaning activities has been demonstrated to be technically practical and efficient. The tank cleaning system has performed as designed and has achieved higher removal efficiencies than initially projected in the PA, as presented in Section A-2 of Appendix A.

Table 6. Percentage of highly radioactive radionuclides removed from all tanks and ancillary equipment.

Radionuclides	Total Ci Generated at INTEC	Residual Ci in Tanks at Closure[a]	Percent Removed at Closure
^{241}Am[b]	9.28E+03	6.97E+00	99.92%
^{137m}Ba[b]	8.95E+06	1.19E+04	99.87%
^{14}C	2.91E–02	3.85E–05	99.87%
^{242}Cm	1.51E+01	1.00E–02	99.93%
^{60}Co	1.67E+03	4.79E–01	99.97%
^{137}Cs[b]	9.46E+06	1.19E+04	99.87%
^{129}I[b]	6.01E+00	5.87E–03	99.90%
^{3}H	7.13E+03	5.43E+00	99.92%
^{94}Nb	1.54E+03	1.60E+00	99.90%
^{59}Ni	3.71E+03	1.90E–01	99.99%
^{63}Ni	4.36E+05	2.17E+01	99.99%
^{237}Np[b]	7.53E+01	3.57E–01	99.53%
^{238}Pu[b]	1.07E+05	9.08E+01	99.92%
^{239}Pu[b]	2.83E+03	2.90E+01	98.98%
^{240}Pu	1.46E+03	1.09E+01	99.25%
^{241}Pu	4.73E+04	1.52E+02	99.68%
^{242}Pu	3.94E+00	7.60E–03	99.81%
^{90}Sr[b]	8.42E+06	6.78E+02	99.99%
^{99}Tc[b]	3.67E+03	5.79E+00	99.84%
^{90}Y[b]	8.42E+06	6.75E+02	99.99%
Total (Ci)[c]	3.59E+07	2.58E+04	99.93%

a. Total Ci at closure includes ^{137m}Ba and ^{90}Y and radionuclide decay to 2012. Based on: (1) heel residuals that are estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points of known height, (2) best estimated radionuclide concentrations from past and recent samples as calculated in the associated engineering design files (Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g, 2005h), and (3) radioactive decay to 2012.

b. Radionuclides that are significant contributors to dose calculations in the 2003 TFF PA (DOE-ID 2003b).

c. Radionuclides shown are contributors to the dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

Of the approximately 36 million Ci in 9.4 million gal of waste generated during spent fuel reprocessing operations, approximately 25,800 Ci are estimated to remain in the tank system following closure, representing about 0.07% of the initial spent fuel waste inventory. Approximately 92% of the total tank waste residual inventory at closure is represented by $^{137}Cs/^{137m}Ba$. The half-life of ^{137}Cs is 30.17 years.

As discussed above, these values have been calculated by decaying the radionuclides to 2012, which is the projected end of closure activities. However, for perspective, if the inventory were to be calculated by decaying the radionuclides to the present (2005), the total Ci remaining at closure would be somewhat higher, on the order of 30,000 Ci. The change is caused by ^{137}Cs and ^{90}S and their daughter products since these radionuclides account for the majority of the radioactivity, and they decay relatively quickly. Calculating these values assuming a decay date to the present results in a slightly higher estimate of total Ci remaining. However, if a table similar to Table 6 was constructed to show percent removed values for current (2005) decay dates, the percent removed values would remain the same, because the relatively short-lived radionuclides would also increase in the INTEC inventory at the same rate as the TFF inventory based on the half-life of each radionuclide.

5.2.2 Cost of Developing a More Efficient Cleaning Technology

In an effort to analyze the cost and benefits to remove the relatively small amount of residual, a cost estimate is provided below. While cleaning began in 2002, development of technology and preparing supporting analysis began earlier. In 1998, a report was prepared that estimated the costs of various options for closure of the TFF (INEEL 1998). The estimate to develop a tank cleaning system without deployment was $46M (not escalated). The estimate was based on a tank cleaning system similar to the system deployed at the TFF. This estimate included the following items and associated costs shown in Table 7.

The expenditure for TFF closure activities from 1999 to 2005 is $35M. The $46M in expenditures to develop a new technology is presented for comparison to the $35M used in this analysis. Seven 300,000-gal tanks and four 30,000-gal tanks have been cleaned from 2002 to 2005. The expenditure of funding for development of a cleaning technology and supporting information to determine the effectiveness of the technology is included in the total amount spent to date.

Table 7. Cost estimate of TFF cleaning system (INEEL 1998).

Activity	Estimate Cost ($M)
Design of System	19.0
Proof of Process	5.8
Site Preparation	9.7
Characterization of Waste	4.6
Tank Isolation	7.0
Total	46.1

Development and deployment of a new technology is estimated to cost slightly more than development and deployment of the existing tank cleaning system. The estimated cost is projected by simply using an escalation rate of 10%. Therefore, the estimate for development and deployment of a new technology is $38.5M. The cost would likely be much higher because the tank cleaning system that has been deployed at the TFF has used much of the existing equipment in the TFF, including the transfer piping (underground shielded piping and valve boxes), the transfer system (steam jets and steam lines), tank access (used existing risers), monitoring system (level detection, alarms, and radiation measurement meters), and the vessel off-gas system. A new system may not use all of the existing equipment. Modification to the TFF, if made for a new technology, would increase costs to deploy the system.

The access to tanks through existing risers is a limiting factor for deployment of a new technology. The risers are a maximum of 46 cm (18-in.) in diameter, with a maximum of five risers on any tank. Existing equipment currently in the risers decreases the available space in risers. The equipment includes instrumentation that cannot be readily removed. If new access must be designed that penetrates the soils, vault roof, and tank it would increase the cost considerably. Additional worker exposure from installation of new risers from tank contents and soils, plus contamination control, would add to costs. Additionally, excavation of soils was not necessary for the tanks cleaned up to 2005 and it is not anticipated for the remaining four tanks. Weight restrictions on the surface of the TFF are also a concern. Multiple pieces of large equipment cannot be placed on the TFF if necessary to deploy a new cleaning technology.

The cleaning technology for the TFF has been designed to use existing systems and commercially available systems or parts. During the review of technologies for cleaning, all known general categories of cleaning technologies were examined. Some types of specific technologies have been designed for tanks at other DOE facilities. Technologies such as a robotic crawler are not viable because of the cooling coils on the bottom of many of the TFF tanks. Therefore, a new technology or a refinement of an existing technology would be developed, designed, tested (found to be more effective), and deployed to the TFF.

Deploying other tank cleaning technologies may delay closure of the TFF past 2012 with associated additional maintenance costs (this is not included in the estimated cost of $38.5M). A factor that also must be considered is that approximately 43% of the inventory in the TFF is not in the tanks but in the sandpads. Therefore, reduction in predicted radiation dose by use of a new tank cleaning system can only decrease to approximately 57% (up to a maximum of 60%) of the total inventory.

Table 8 shows example efficiencies of a hypothetical tank cleaning technology up to a maximum of 60%. Also shown are the projected dose in mrem/yr of the all-pathways assessment, and the associated reduction in dose in mrem/yr. Table 8 also shows the cost per mrem of deploying a new technology. Even using the most efficient system (which cleans the tanks completely), the dose reduction for a 50-year period (50 years of exposure) is approximately 0.27 mrem/yr, and the cost per mrem reduction is approximately $2.8M.

Table 8. All-pathways dose reduction and cost per mrem.

	Efficiency of Technology (%)			
	20	40	50	60
Dose (mrem/yr)	0.368	0.276	0.23	0.184
Dose reduction (mrem/yr)	0.092	0.184	0.23	0.276
Cost/mrem reduction	$8,369,565.22	$4,184,782.61	$3,347,826.09	$2,789,855.07

Note: See Table 19, which presents the scaled groundwater all-pathways dose (0.46 mrem/yr).

The system design and testing that was used in this comparison was estimated at $38.5M. Based on other estimates (INEEL 1998), the cost could be approximately $50M if a 10% escalation was added.

5.2.3 Conclusions

Based on specific conditions, the INL Site determined that a limited number of technologies would effectively remove waste residuals from the tanks. The benefit of developing additional technologies and even complete tank removal was evaluated for reducing waste residuals in the TFF. It is likely a system could be developed and deployed to remove the remaining residual by other remote means (INEEL 1998). However, initiating a long-term development project to develop other technologies would take many years to complete, and the project would be very costly (INEEL 1998). Although the cleaning to date has been successful, the small volume of residual certainly poses a direct radiation concern for anyone who would be in close proximity to the residual. No "hands-on" cleaning is practical and any new technology may require workers to contact the residual waste so worker exposure would increase while the projected dose to the public from the residual waste, which is already low, would be lowered only slightly by using the new technology. In practice, during tank cleaning operations, even when the radiation monitor indicates that radioactivity levels are no longer decreasing and the videotape shows that the tank is clean, tank washing is continued for another day by flushing with several thousand gallons of water. Any appreciable decrease in the projected radiation dose to the public would be gained only if the TFF tank system is removed entirely.

Tank cleaning worker exposure is estimated to total approximately 650 mrem per tank for 23 workers (30 mrem per worker), which results in a total exposure of about 7.15 rem for 11 300,000-gal tanks (Appendix B-5; Jacobson 2002). Tank removal worker doses are significant because the action would require a large project to excavate the tanks, cut them up, and package them for disposal. Worker exposure is estimated to be 1,070 mrem/yr/worker for an average of 326 workers/yr for an estimated 26 years for a total exposure of over 9,000 rem (Appendix B-5; INEEL 1998). Therefore, this option would be several billion dollars (INEEL 1998), cause more exposure, and result in radioactive waste for which disposal would be difficult or not possible.

As shown in Section 7, the remaining residuals, if left in place, pose a potential radiation dose to a member of the public on the order of 0.5 mrem/yr, and approximately half of this radiation dose is due to residuals in the tanks. Therefore, any new tank cleaning system could only achieve a reduction of approximately 0.25 mrem/yr, regardless of cost. With typical average doses to the public from natural sources and medical treatment in the range of 300–400 mrem/yr, it is not judged practical or cost effective to reduce the estimated dose from TFF by such a small amount.

6. RADIONUCLIDE CONCENTRATIONS OF STABILIZED WASTE

Key Points

- The waste concentration in the grouted 300,000-gal tanks is not reduced by the amount of additional residual waste that may be removed from the cleaned 300,000-gal tanks during grouting operations.
- The same degree of radionuclide removal is expected for the four 300,000-gal tanks remaining to be cleaned as that achieved in the seven tanks that have been cleaned, for the reasons explained previously in this draft 3116 Determination.
- Calculations and analyses assume that the TFF will be closed in 2012 as planned.
- Concentrations of waste solidified in grout are calculated for the closed tanks, vaults, and piping, and shown along with the Class C concentration limits in 10 CFR 61.55.

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
>
> (3)(A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (3)(B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations, and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and
>
> (iii) pursuant to plans developed by the Secretary in consultation with the Commission.

Section 3116(a)(3) provides that, regardless of whether the waste exceeds or does not exceed the concentration limits for Class C LLW as set out in 10 CFR 61.55, the Secretary must determine that the waste will be disposed of in compliance with the performance objectives of 10 CFR 61, Subpart C and that the waste will be disposed of in accordance with State-approved closure plans. In Section 7 of this document, information is presented that demonstrates that the waste will be disposed of in compliance with the performance objectives of 10 CFR 61, Subpart C. In Section 8 of this document, information is presented that demonstrates that waste will be disposed of in accordance with State-approved closure plans.

In situations where the waste is determined to exceed the concentration limits for Class C LLW, Section 3116(a)(3)(B)(iii) provides for consultations with NRC about the disposal plans for the waste. The DOE is not deciding in this draft 3116 Determination whether the waste does or does not exceed the concentration limits for Class C LLW since there is no clearly applicable NRC guidance on applying the concentration limits set out in 10 CFR 61.55 to situations like the TFF tank system. Nevertheless, DOE intends to take full advantage of the consultation process established by Section 3116 and is requesting that NRC identify what changes, if any, it would recommend to DOE's disposal plans (i.e., the actions DOE intends to undertake to dispose of the waste, as described in this draft 3116 Determination) to meet the criteria of Section 3116.

In order to fully comply with the provisions of Section 3116 and to facilitate the consultations with NRC, this section discusses radionuclide concentrations of wastes solidified in grout in the TFF tank system, including the tanks, vaults, and piping. This discussion is structured in a manner that permits a comparison of various levels of solidified waste.

Grout will be used to stabilize the waste. Grout mix designs have been developed for effective closure of the TFF tanks. Appendix C presents the preliminary grout mix design and quality assurance (QA) tests, as well as results from grout testing performed thus far. Measures will be instituted to ensure delivery of a consistent grout.

In the following subsections, the concentrations of residuals in the primary TFF components are calculated and shown along with the 10 CFR 61.55 Class C concentration limits. For clarity, the waste concentrations in the 300,000-gal tanks, 30,000-gal tanks, and piping are calculated and discussed separately; the waste concentrations in the vaults and sandpads are calculated and discussed together.

6.1 Waste Concentration in the Grouted 300,000-gal Tanks

Comprehensive mockup testing (INEEL 1999b) and engineering considerations indicate that an estimated 225 m^3 of grout in each 300,000-gal tank is necessary to provide a level grout fill over the placement locations. The mockup testing shows that strategic pours of grout may provide an ability to remove some portion of any residual waste remaining after water cleaning is complete. Each of the grout placement locations described in Section 2 are used to move waste residual toward the steam jets for removal. The grout placement sequence to move remaining solids and liquid toward the removal pumps is estimated to use 85 m^3 of grout. A final pour of 140 m^3 of grout is needed to stabilize any residuals that appear on the surface after the first grout placements (EDF-2973, 2001). The total volume of grout for this operation is 225 m^3, which results in a layer of grout about 1.2 m (4 ft) thick from the tank floor. In addition to this grout layer, the 300,000-gal tanks will be filled completely with grout, including the tank domes, to provide for long-term stability of the tank structure.

In calculating the radionuclide concentration, 10 CFR 61.55(a)(8) states that the radioactivity in the waste may be divided by the volume of the final waste form. Therefore, to calculate a waste concentration in these tanks, the estimate of final residual waste inventory at closure discussed in Section 2 is divided

by the volume of the final waste form. Videos and photographs of the tank walls show staining and discoloration, and no discernible buildup of waste residuals. Therefore, no source term for the tank walls was included in the tank inventory discussed in Section 2.

After cleaning is completed on each tank, the radionuclide inventory in the 300,000-gal tanks is in a dispersed layer across the tank bottom. The layer varies from 0.32 cm (0.13 in.) in thickness to near zero and is not concentrated in one specific location in the tank. The addition of grout for final closure is not expected to significantly mix with the residual waste or concentrate the radionuclide inventory. Mockup testing of grout placements (INEEL 1999b) in the tanks has shown that the resulting waste form will not be completely homogeneous; rather, the waste form will be a grouted monolith with some of the remaining residuals mixed in with the grout, some trapped between grout layers, and some encapsulated between the tank structure and the grout.[ee]

Table 9 shows the calculations of the residual waste inventory at closure for Tank WM-182 averaged over 225 m^3 of grout (with a density of 2.1 g/cm^3).[ff] For reference, the Class C concentration limits for long- and short-lived radionuclides (Tables 1 and 2 of 10 CFR 61.55) are shown in Tables 9 and 10. Since the tank is considered part of the final waste form, the mass of the stainless steel tank is included in the calculation of average radionuclide concentrations shown in Tables 9 and 10. Only the stainless steel mass, which encloses the engineering grout pour, is applied to the calculation.

Tank WM-182 inventory estimates are presented because this tank contains the highest concentrations of radionuclides of the cleaned tanks. Inventories at closure for all tanks are presented in Appendix A. The sum-of-the-fractions results are shown for the Tank WM-182 grouted waste form. The other 300,000-gal tanks that have been cleaned have a lower residual waste inventory at closure than Tank WM-182. As noted in Subsection 2.4.2, the residual waste inventory at closure for the 300,000-gal tanks does not take credit for any additional residuals that may be removed during grouting operations. Therefore, the comparisons shown in Tables 9 and 10 are also made without accounting for this additional residual waste removal.

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ee. Nevertheless, the resulting waste form may be considered "reasonably homogeneous," as discussed in the *Branch Technical Position on Concentration Averaging and Encapsulation* (NRC 1995). The Branch Technical Position describes a reasonably homogeneous waste type as "one in which the radionuclide concentrations are likely to approach uniformity in the context of the intruder scenarios." The TFF PA intruder scenarios include activities such as excavation into TFF piping and drilling into grouted tanks. The remaining waste residuals in the closed tanks would not be concentrated in any specific location in the tank. If such activities were to occur, they would, by their nature, result in a significant amount of additional mixing of grout, surrounding soils, and a relatively small volume of waste residuals. Therefore, the resulting inventory and dose would be similar regardless of where the drilling or excavations occurred within a specific tank. In this context, the TFF tanks will meet this definition of reasonably homogeneous when the residuals are stabilized with grout and the radionuclide concentrations of such a mixture would likely approach "uniformity."

ff. The NRC provides guidance on calculating final waste form concentrations when the waste is not mixed well with grout in its *Branch Technical Position on Concentration Averaging and Encapsulation* (NRC 1995). However, this Branch Technical Position provides general guidance for concentration averaging and does not establish a standard method(s) for concentration averaging for tanks. In particular, residual waste in the bottom of a tank is not addressed specifically, but the underlying principles of the guidance suggest that averaging the radioactivity of the waste over the volume of grout used to stabilize the waste may be appropriate. For the 300,000-gal tanks, this volume is 225 m^3.

Table 9. Radionuclide concentrations in the final Tank WM-182 grouted waste form (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	Tank Inventory (Ci)[b]	Tank Inventory in Ci/m^3	Tank Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	4.2E-01		8.5E-01	100	0.0085
^{14}C	***5.7E+03***	***5.0E–06***	***2.2E–08***		***8***	***0.0000000027***
^{242}Cm	4.5E–01	1.3E–03		2.6E–03	20,000	0.00000013
^{129}I	***1.6E+07***	***7.7E–04***	***3.4E–06***		***0.08***	***0.000043***
^{94}Nb	***2.0E+04***	***2.1E–01***	***9.1E–04***		***0.2***	***0.0045***
^{59}Ni	***7.5E+04***	***2.5E–02***	***1.1E–04***		***220***	***0.00000050***
^{237}Np	2.1E+06	4.7E–02		9.4E–02	100	0.00094
^{238}Pu	8.8E+01	1.1E+01		2.3E+01	100	0.23
^{239}Pu	2.4E+04	3.4E+00		6.8E+00	100	0.068
^{240}Pu	7.0E+03	1.4E+00		2.7E+00	100	0.027
^{241}Pu	1.4E+01	1.9E+01		3.9E+01	3,500	0.011
^{242}Pu	3.8E+05	9.9E–04		2.0E–03	100	0.000020
^{99}Tc	***2.1E+05***	***7.6E–01***	***3.4E–03***		***3***	***0.0011***
Sum of the Fractions						**0.35**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining nuclides are concentration limits in units of nCi/g.
b. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.
c. Table 1 of 10 CFR 61.55.

Table 10. Radionuclide concentrations in the final Tank WM-182 grouted waste form (Table 2 of 10 CFR 61.55).

Radionuclide	Half-Life (yr)	Tank Inventory (Ci)[a]	Tank Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.1E+03	5.0E+00	4,600	0.0011
^{63}Ni	1.0E+02	2.9E+00	1.3E–02	700	0.000018
^{90}Sr	2.9E+01	2.3E+01	1.0E–01	7,000	0.000015
Sum of the Fractions					**0.0011**

a. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.
b. Table 2 of 10 CFR 61.55.

As discussed previously, the INL Site intends to fill the 300,000-gal tanks to capacity with grout. The 2.1-g/cm^3 density of the grout is similar to the density of the native soils that had been originally excavated for the installation of the TFF (WINCO 1992). The addition of grout will fill voids to prevent subsidence but will not result in a significant increased load on the basalt, which forms a base for the TFF tanks. Some of the 300,000-gal tanks have cooling coils in the bottom and on the walls of the tanks, but they do not contain significant vertical structures. Figure 14 is a schematic of the grout-filled tank. For perspective, the figure shows the results of averaging the estimated remaining radioactivity with various levels of grout, including the full tank volume (approximately 1,500 m^3). Based on mockup testing, a depth of approximately 1.2 m (4 ft) is necessary to provide for immobilization of the residual on the tank

floor (225 m^3). This depth of grout is referred to as the engineering grout pour (225 m^3), and the volume at the height of grout where the average concentration of the grouted vault would not exceed Class C concentration limits (approximately 80 m^3) is referred to as the level of grout that meets Class C.

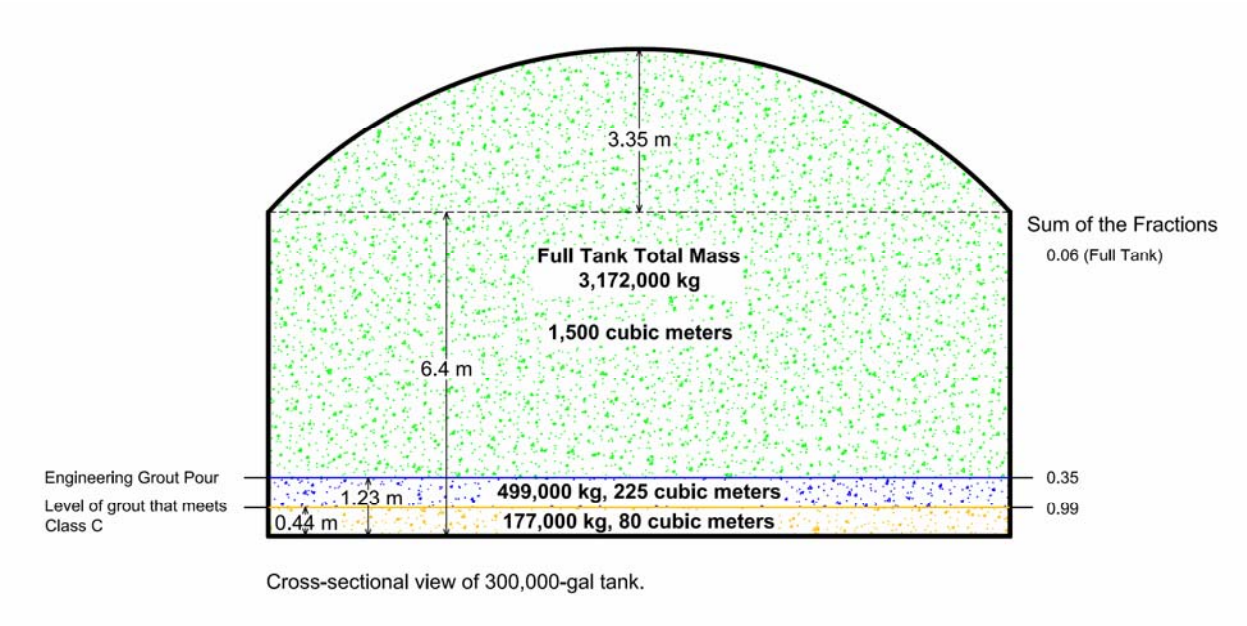


Figure 14. Schematic of a grout-filled tank showing the tank volume, the mass of grout, and the sum of the fractions for each volume or mass of grout.

The grouted waste residuals in the stainless steel tanks provide a solid physical form, as required by 10 CFR 61.55, which cross-references 10 CFR 61.56. In addition, the tank, vaults, and vault grout (for the 300,000-gal tanks) provide structural stability for the waste and help prevent the dispersal of the radionuclides into the surrounding environment. For the 300,000-gal tanks, the vault grout will fill the vault area, incorporating any waste residuals in the vaults into a solid matrix. Characteristics of the tank and vault grout are presented in Appendix C, along with the required QA tests and results from grout testing.

6.2 Waste Concentration in the Grouted 30,000-gal Tanks

The residual waste remaining in each 30,000-gal tank contains approximately 36 Ci, which is approximately 1.5% of the residual waste inventory in a typical 300,000-gal tank. As discussed in Section 2, the residual waste inventory at closure for the 30,000-gal tanks assumes that each tank has a 5-mil (0.005-in.) thick film on the lower half of the tanks. The 5-mil (0.005-in.) thickness was used as a conservative estimate since solid samples were not collected because of a lack of material to sample. The tanks had not contained acidic waste for at least 20 years, and the heat from the steam valve condensate allowed the development of what appeared to be a biological film on the tanks. However, this film could not be sampled to determine the radionuclide concentrations. Therefore, conservative thickness and radionuclide concentrations assumptions are used. For the radionuclide concentrations in this film layer, the analytical results from the solid samples from Tank WM-183 in Ci/kg are applied to this mass of solid material. The liquid residuals were sampled and analyzed as discussed in Section 2. Liquid sampling results averaged are approximately 0.23 Ci in liquids with 36.1 Ci of solid residual.

During grouting operations, the tanks will be filled to the midway point initially because the thin film observed in the tanks does not extend beyond the bottom half of the tank. This grout pour will be performed to contain the residual on the tank wall. These tanks will be filled completely with grout prior to closure. Since the bottom area of a horizontal, cylindrical tank is relatively small compared to a flat-bottomed tank, and minimal residuals remain in the tank bottom, the multi-point grout placement technique planned for the 300,000-gal tanks will not be necessary for these tanks.

The volume of grout needed to fill the tanks to the half of the volume is 58 m^3 with a density of 2.1 g/cm^3. Since the tank is considered to be part of the final waste form, the mass of the stainless steel tank is included in the concentration averaging calculation. Only stainless steel mass that encloses the engineering grout pour (half of the tank) is applied to the calculation. Tables 11 and 12 show the calculations of the residual waste inventory at closure for a 30,000-gal tank averaged over 58 m^3 of grout (with a density of 2.1 g/cm^3). For reference, the Class C concentration limits for long- and short-lived radionuclides (Tables 1 and 2 of 10 CFR 61.55, respectively) are also shown.

Table 11. Radionuclide concentrations in a final 30,000-gal tank grouted waste form (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	Average 30,000-gal Tank Inventory (Ci)[b]	Average 30,000-gal Tank Inventory in Ci/m^3	Average 30,000-gal Tank Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	6.4E–03		4.9E–02	100	0.00049
^{14}C	***5.7E+03***	***1.1E–07***	***1.9E–09***		***8***	***0.00000000024***
^{242}Cm	4.5E–01	2.0E–05		1.5E–04	20,000	0.0000000077
^{129}I	***1.6E+07***	***1.2E–05***	***2.0E–07***		***0.08***	***0.0000025***
^{94}Nb	***2.0E+04***	***3.1E–03***	***5.3E–05***		***0.2***	***0.00027***
^{59}Ni	***7.5E+04***	***3.8E–04***	***6.5E–06***		***220***	***0.000000030***
^{237}Np	2.1E+06	7.1E–04		5.5E–03	100	0.000055
^{238}Pu	8.8E+01	1.7E–01		1.3E+00	100	0.013
^{239}Pu	2.4E+04	5.1E–02		4.0E–01	100	0.0040
^{240}Pu	7.0E+03	2.0E–02		1.6E–01	100	0.0016
^{241}Pu	1.4E+01	2.9E–01		2.3E+00	3,500	0.00065
^{242}Pu	3.8E+05	1.5E–05		1.1E–04	100	0.0000011
^{99}Tc	***2.1E+05***	***1.2E–02***	***2.0E–04***		***3***	***0.000066***
Sum of the Fractions						**0.020**

a. Radionuclides in ***bold italics*** are concentration limits in units of Ci/m^3; remaining nuclides are concentration limits in units of nCi/g.
b. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.
c. Table 1 of 10 CFR 61.55.

Table 12. Radionuclide concentrations in a final 30,000-gal tank grouted waste form (Table 2 of 10 CFR 61.55).

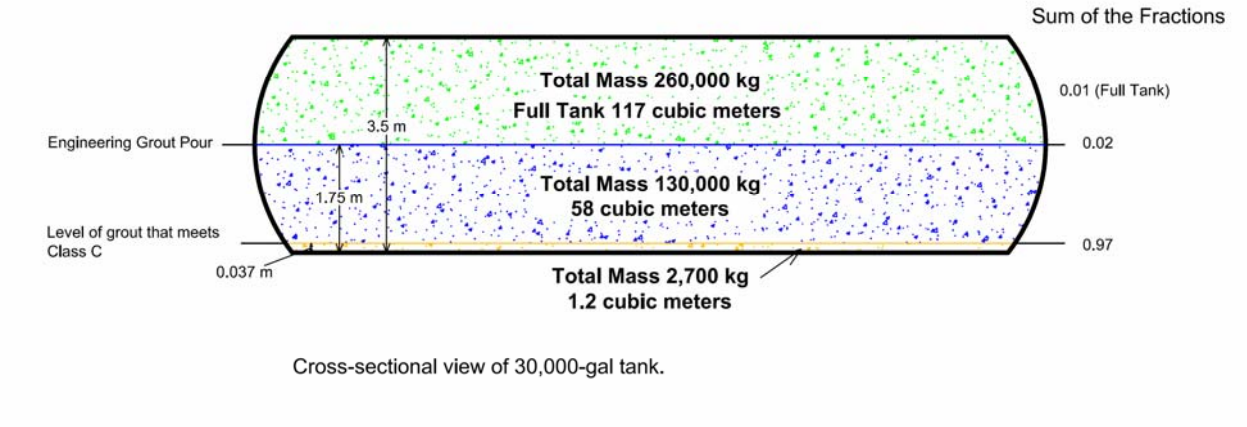
Radionuclide[a]	Half-Life (yr)	Average 30,000-gal Tank Inventory (Ci)[a]	Average 30,000-gal Tank Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.7E+01[c]	3.0E–01	4,600	0.00006
^{63}Ni	1.0E+02	4.3E–02	7.4E–04	700	0.0000011
^{90}Sr	2.9E+01	4.5E–01[c]	7.8E–03	7,000	0.0000009
Sum of the Fractions					**0.00007**

a. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.

b. Table 2 of 10 CFR 61.55.

c. Radioactivity for ^{137}Cs (^{137}Ba) and ^{90}Sr (^{90}Y) daughters are not shown in this table.

Like the 300,000-gal tanks, the 30,000-gal tanks will be filled to capacity with grout at final closure. This action will ensure that the tank will be stabilized and reduce subsidence. For perspective, Figure 15 shows the grout-filled 30,000-gal tanks, the mass of the grout at three levels, and the sum of the fractions associated with those points.



Cross-sectional view of 30,000-gal tank.

Figure 15. Schematic of a grout-filled 30,000-gal tank showing the tank volume, the mass of grout, and the sum of the fractions.

6.3 Waste Concentration in the Grouted 300,000-gal Tank Vaults and Sandpads

The tank secondary containment system consists of the tank vaults and sandpads, and is discussed together in this subsection. An inventory has not been developed for the tank vaults because solids are not present in the vaults and liquid is confined to the volume of the sump. Generally, the sumps are 0.3- by 0.3-m (1- by 1-ft) concrete enclosures in the vault bottom. Some of the sumps are slightly larger. Any residual waste inventory at closure for the vaults is considered negligible when compared to the sandpad inventory.

For the vaults around the 300,000-gal tanks, contamination of the vault surfaces is from liquid releases from valves boxes or directly from the tanks. As discussed in Section 2, the tanks have never leaked into the vaults. Waste entered the vaults directly from the tanks only twice from Tanks WM-185 and WM-187 during back-siphoning events. Typically, vaults have contained contaminants from only two sources, secondary containment of process piping and valve boxes. Process piping and valve leaks are infrequent occurrences. In the event of a leak, the system of liquid level indicators and radiation detection instruments would alert operators of a leak and the contamination would be removed well before the volume of the vault sump (volume is approximately 0.3 m^3) was exceeded. Therefore, such sources contributed only minor amounts of contamination. Additionally, vaults collected water from precipitation events or snowmelt. This water entered the vault from the vault top, which is made of pre-stressed concrete slabs that were never designed to prevent water inflow. Vaults associated with the cleaned tanks have been flushed and sampled. Solids were not expected in the tank vaults and solid samples were not collected. Solids were not expected because waste was not contained in any vault for a long period of time. Vaults for Tanks WM-185 and WM-187 did hold several thousand gallons of waste from a back-siphoning event in each vault in 1962. The waste was removed from each vault within 24 hours after each event began.

During recent cleaning operations, the tank vaults were flushed with water during the cleaning process, but access to the vaults did not allow for the aggressive cleaning and agitation that was performed in the tanks. Sampling from the vault sumps may have contributed to elevated radionuclide concentrations in liquid samples. The vault sumps are the low points in the vaults designed to capture leaks from valve boxes and to allow removal of infiltration from precipitation and waste as necessary. The vault sumps were never emptied completely because the steam jets are approximately 5 cm (2 in.) from the sump floor. Therefore, residual contamination in the sump may have biased the sample concentrations. No solids have been observed in the tank vaults other than debris. Liquids do not collect in the vault because the floors are sloped toward vault sumps. After cleaning, the water used to flush the vaults was pumped out using the steam jets in vault sumps. The sumps are generally 0.3 m (1 ft) square (some vaults have larger sumps). Review of the sampling data shows that the concentrations of radionuclides in liquid vault samples do not exceed the liquid concentration of the cleaned tanks.

A conservative engineering analysis is used to estimate the contamination remaining in the sandpads because sampling the sandpads is not technically feasible and no sandpad sampling data exist. Since the sandpads are located directly beneath the tanks, any useful sampling would require a task equal to complete tank removal given the limited access. The engineering analysis indicates radionuclides from the early contamination events remain absorbed to the sand despite multiple washings (DOE-ID 2003b). Cleaning the sandpad is not practical given the limited access and location of this sand. The

commercial-grade sand in the sandpads is in solid physical form and will be surrounded by the vault, the grout, and the tank (filled with grout).[gg]

The vault will be filled completely with grout. Incremental steps in the grout fill will accomplish covering the sandpad by making grout pours through the two available risers, filling to level the initial grout pour, and finally filling the entire vault. A minimum grout height of 1 m (3 ft) is necessary to allow the grout to flow around the vault and rise to a level of at least 46 cm (18 in.) above the grout floor. The 46-cm (18-in.) height is desirable because the sandpad and associated dike are 15 cm (6 in.) above the floor and approximately 46 cm (18 in.) level will ensure encapsulation of the sandpad. Grout poured into the vault will fill the area surrounding the sandpads, leaving no voids and providing structural integrity to the sandpads. Liquid entering the vault system will be drained from the vault sump prior to closure. Grouting of the vault area surrounding the sandpad will incorporate any remaining liquids. The inventories of the vaults are insignificant when compared to the sandpad because very little liquid remains in the vault sumps and no solid residual is present.

The residual waste inventory at closure for the sandpads was presented in Section 2. The calculations in Tables 13 and 14 are based on a total sandpad volume of 23.39 m^3 (Staiger 1999) and the mass of the sandpad and vault grout (65 m^3 or 1 m [3 ft] high in a typical vault) of 136,500 g. Tables 13 and 14 show the radionuclide concentrations in the grouted vaults. For reference, the Class C concentration limits for long- and short-lived radionuclides (Tables 1 and 2 of 10 CFR 61.55) are also shown.

The radionuclide ^{59}Ni is not included in the sandpad inventory. The initial amount of each radionuclide in the tanks that was back-siphoned to the vault and contacted the sandpad at the time of the accidental spills was evaluated using limited analytical data and ORIGEN2. The ORIGEN2 data were corrected to the ^{137}Cs concentration in Tank WM-185 one month before the incident. Tank WM-185 was used because of its slightly higher ^{137}Cs concentration than in Tank WM-187. However, this ORIGEN2 modeling did not include the activation product ^{59}Ni. The concentration of ^{59}Ni in the sandpads will not affect the comparisons made in Tables 13 and 14 because ^{59}Ni is typically found in very low concentrations as evidenced by the concentrations in the tanks and piping.

gg. The sandpads are part of the tank ancillary equipment and are not considered to be contaminated media outside of the tank system.

Table 13. Radionuclide concentrations in the grouted vaults (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	Sandpad Inventory (Ci)[b]	Sandpad Inventory in Ci/m^3	Sandpad Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	1.9E+00		1.4E+01	100	0.14
^{14}C	***5.7E+03***	***3.9E–07***	***6.0E–09***		***8***	***0.00000000075***
^{242}Cm	4.5E–01	1.4E–05		1.0E–04	20,000	0.0000000051
^{129}I	***1.6E+07***	***1.1E–06***	***1.7E–08***		***0.08***	***0.00000021***
^{94}Nb	***2.0E+04***	***2.3E–02***	***3.5E–04***		***0.2***	***0.0018***
^{237}Np	2.1E+06	3.7E–04		2.7E–03	100	0.000027
^{238}Pu	8.8E+01	2.1E+00		1.5E+01	100	0.15
^{239}Pu	2.4E+04	1.6E+00		1.2E+01	100	0.12
^{240}Pu	7.0E+03	3.5E–01		2.6E+00	100	0.026
^{241}Pu	1.4E+01	2.3E+00		1.7E+01	3,500	0.0048
^{242}Pu	3.8E+05	5.7E–05		4.2E–04	100	0.0000042
^{99}Tc	***2.1E+05***	***2.0E–12***	***2.1E–15***		***3***	***0.00000000000000071***
Sum of the Fractions						**0.44**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining nuclides are concentration limits in units of nCi/g.

b. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.

c. Table 1 of 10 CFR 61.55.

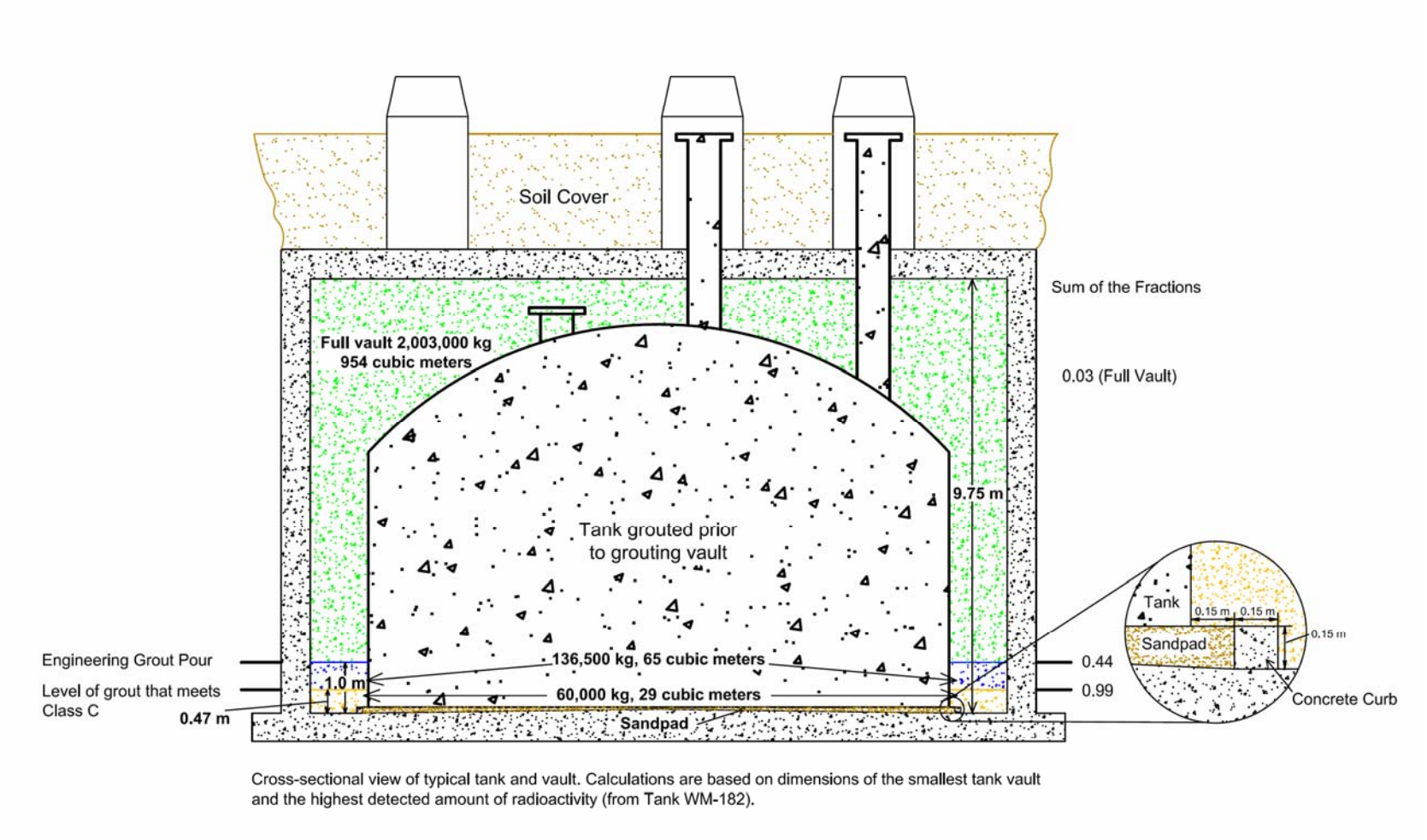
Table 14. Radionuclide concentrations in the (Table 2 of 10 CFR 61.55) grouted vaults (DOE-ID 2003b).

Radionuclide	Half-Life (yr)	Sandpad Inventory (Ci)[a]	Sandpad Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.6E+03	2.5E+01	4,600	0.00543
^{63}Ni	1.0E+02	1.7E-10	2.6E-12	700	3.71E-15
^{90}Sr	2.9E+01	2.5E+02	3.8E+00	7,000	0.000543
Sum of the Fractions					**0.00597**

a. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.

b. Table 2 of 10 CFR 61.55.

Engineering judgment and an analysis that determined the depth of grout that should be poured in vaults (1 m [3 ft]) to help prevent excessive grout cracking and shrinkage were prepared for a tank closure report (INEEL 1998). The depth of grout that meets the Class C concentration limits (level of grout that meets Class C) and the engineering grout pour are also delineated in Figure 16. Therefore, the requirement that the waste be incorporated into a solid physical form (for stability and to prevent waste migration) is met for the sandpads. Figure 16 shows the grout-filled vault including the sandpad, the sum of the fractions at the incremental steps of filling, and the associated mass at each step.



Cross-sectional view of typical tank and vault. Calculations are based on dimensions of the smallest tank vault and the highest detected amount of radioactivity (from Tank WM-182).

Figure 16. Schematic of a grout-filled vault showing the vault volume, the mass of grout, and the sum of the fractions.

6.4 Waste Concentration in the Grouted Piping, Encasements, and Valve Boxes

Like the other TFF system components, the TFF ancillary SSCs (e.g., piping, encasements, valves, and valve boxes) will be grouted at final closure (i.e., disposal). The grout will provide a solid physical form to incorporate any waste residuals and to prevent their dispersal into the surrounding environment. The piping and SSC grout must necessarily be able to flow more easily during grouting operations, and consequently, this grout will have a higher water content than the grout mixture used for other TFF system components. However, the grout will still be able to harden into a solid physical form to provide waste stability. The pour process will be adjusted so that effective grout distribution through the piping, encasements, and other equipment is achieved. Characteristics of the grout are presented in Appendix C, along with the required QA tests and results from grout testing.

To develop waste concentrations in the grouted piping, the residual waste inventory at closure for the piping presented in Section 2 (15.5 kg of SBW solids) is averaged over the 3,231 linear m (10,600 linear ft) of 5-cm (2-in.) piping (EDF-2973, 2001). Tables 15 and 16 show the radionuclide concentrations of the grouted piping. For reference, the Class C concentration limits for the long- and short-lived radionuclides are also shown. For perspective, Tables A-13 and A-14 in Appendix A show the radionuclide concentrations of the piping without adding grout to the pipe. In this case, the waste form is assumed to be Schedule 40 stainless steel pipe and the associated radioactive contamination. Tables 15 and 16 include the mass and volume of the stainless steel piping and the mass and volume of the grout in the calculations while Tables A-13 and A-14 include only the mass and volume of the piping. The entire mass of stainless steel piping is included in the calculations.

The valve boxes and encasements contain significantly less contamination after cleaning than the process piping (as described in Subsection 6.3). Therefore, the radionuclide concentrations in the valve boxes and encasements after closure would be less than those shown for the piping.

Table 15. Radionuclide concentrations in the piping (with grout) (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	Piping Inventory (Ci)[b]	Piping Inventory in Ci/m^3	Piping Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	5.3E–03		1.7E–01	100	0.0017
^{14}C	***5.7E+03***	***6.2E–08***	***7.1E–09***		***8***	***0.00000000089***
^{242}Cm	4.5E–01	1.7E–05		5.3E–04	20,000	0.000000026
^{129}I	***1.6E+07***	***9.7E–06***	***1.1E–06***		***0.08***	***0.000014***
^{94}Nb	***2.0E+04***	***2.6E–03***	***2.9E–04***		***0.2***	***0.0015***
^{59}Ni	***7.5E+04***	***3.1E–04***	***3.6E–05***		***220***	***0.00000016***
^{237}Np	2.1E+06	5.9E–04		1.9E–02	100	0.00019
^{238}Pu	8.8E+01	1.4E–01		4.6E+00	100	0.046
^{239}Pu	2.4E+04	4.3E–02		1.4E+00	100	0.014
^{240}Pu	7.0E+03	1.7E–02		5.4E–01	100	0.0054
^{241}Pu	1.4E+01	2.4E–01		7.8E+00	3,500	0.0022
^{244}Pu	3.8E+05	1.2E–05		4.0E–04	100	0.0000040
^{99}Tc	***2.1E+05***	***9.6E–03***	***1.1E–03***		***3***	***0.00036***
Sum of the Fractions						**0.071**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining nuclides are concentration limits in units of nCi/g.
b. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.
c. Table 1 of 10 CFR 61.55.

Table 16. Radionuclide concentrations in the piping (with grout) (Table 2 of 10 CFR 61.55).

Radionuclide	Half-Life (yr)	Piping Inventory (Ci)[a]	Piping Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.4E+01	1.6E+00	4,600	0.00036
^{63}Ni	1.0E+02	3.6E–02	4.1E–03	700	0.0000059
^{90}Sr	2.9E+01	2.9E–01	3.4E–02	7,000	0.0000048
Sum of the Fractions					**0.00037**

a. Radioactive decay to 2012; the sum of the fractions will not significantly change if decayed to 2005.

b. Table 2 of 10 CFR 61.55.

7. WASTE WILL BE DISPOSED OF IN ACCORDANCE WITH PERFORMANCE OBJECTIVES IN 10 CFR 61, SUBPART C

Key Points

- The TFF PA (DOE-ID 2003b) was prepared in accordance with DOE guidance. Moreover, the information in the PA demonstrates that the performance objectives in 10 CFR 61, Subpart C are met. The PA is used to demonstrate that the TFF residual waste will be disposed of in accordance with these performance objectives as required by Section 3116 of the NDAA.
- Grout in the tanks and tank vaults provides structural stability to the waste form and prevents collapse.
- The PA performance period is 1,000 years after TFF closure. The intruder scenarios produce the greatest dose in the first 100 years after closure. The groundwater pathway was examined up to 1 million years post-closure, as part of the sensitivity analysis. Therefore, the effective period of performance complies with NRC guidance of 10,000 years.
- The institutional control period is 100 years following closure (2012–2112). The post-institutional control period is 2112–3012.
- Members of the public are assumed to reside 100 m (328 ft) from the TFF during the post-institutional control period. The all-pathways dose assessment used the well location where the highest concentration of radionuclides was predicted in the aquifer (600 m [1,969 ft] from TFF).
- Grouted tanks and piping are assumed to fail (release contaminants) at 500 years post-closure. Grouted vaults are assumed to fail (release contaminants) at 100 years post-closure.
- Inadvertent intruder scenario assumptions:
  - Institutional controls are effective for at least 100 years following closure.
  - Only the intruder-drilling and post-drilling scenarios are applicable for the tank and sandpad contamination as discussed in Subsection 7.2. The intruder-construction and intruder post-construction scenarios are considered for the piping inventory located less than 3 m (10 ft) below the surface.

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
>
> > (3)(A)(i) [will be disposed of] in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations.

Title 10 CFR 61, Subpart C, Sections 61.40 through 61.44, detail performance objectives the NRC established for land disposal of radioactive waste. For LLW disposal, 10 CFR 61.40, "General Requirement," states "land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives §§61.41 through 61.44." These sections address protection of the general

population from radioactivity releases, protection of individuals from inadvertent intrusion onto the disposal site, protection of individuals during disposal facility operations, and disposal site stability after closure. Compliance with requirements in 10 CFR 61.40–61.44 is discussed in Subsections 7.9–7.13, respectively.

This section documents that the TFF PA shows that disposal of the waste residuals and TFF tank system will meet 10 CFR 61, Subpart C performance objectives. The PA was developed originally as part of a DOE Tier 1 closure plan (DOE-ID 2003c) and as the support for a DOE draft waste determination prior to enactment of Section 3116 of the NDAA. As such, the PA was prepared in accordance with DOE guidance. However, the information in the PA demonstrates that the LLW land disposal performance objectives of 10 CFR 61, Subpart C are met.

The site-specific *Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory* (DOE-ID 2003b) considers all the sources of radioactive waste that may remain in the closed TFF, including the tank residuals, tanks, vaults, sandpads, piping, and other ancillary components for the 11 300,000-gal tanks and vaults and the four 30,000-gal stainless steel tanks. The assessment calculates potential releases to the public, including those that result from inadvertent intrusion, to provide a reasonable expectation that the performance objectives are not exceeded for a 1,000-year period after closure, which is planned to be in 2012. The PA uses the current or planned land use as a basis for scenario development. The receptor is placed in all scenarios where the estimated radiation dose from the TFF is greatest. Therefore, the exposure date and location are placed at the predicted maximum radiation dose. Assumptions in the PA include grouting all voids such that the tanks and tank vaults (for the 300,000-gal tanks) are filled with grout to provide structural stability to the waste form and prevent collapse. The PA process uses conceptual models that link radionuclide inventory, release (or source term), environmental transfer, and impact assessment. The final step is determining radiological doses to receptors. These analyses are used in this draft 3116 Determination to demonstrate the TFF closure (i.e., disposal) is in compliance with Section 3116 criteria.

The potential dose posed by the TFF closure (i.e., disposal) is assessed by conservatively estimating the release and transport of radionuclides from (1) the release of radionuclides from the tanks, sandpads, and piping, and subsequent transport through the environment; and (2) exposures to members of the public via air, groundwater, and food chain pathways. Two receptor types are assessed: a member of the public and an intruder. During the operational and institutional control periods, a member of the public is assumed to reside at the INL Site boundary. During the post-institutional control period (100 years after closure), a member of the public is assumed to reside at the INTEC facility boundary. This receptor is used to evaluate the drinking water pathway, all-pathways (including food chains), and air pathway doses for comparison to the performance objectives. The hypothetical intruder receptor type is assumed to intrude inadvertently onto the TFF during the post-institutional control period under four scenarios.

Calculations in sensitivity and uncertainty analyses for all analyses other than the inadvertent intruder analysis were carried out beyond 1,000 years to determine the maximum impact regardless of the time at which the maximum occurs. Thus, the timeframe for the analysis should be long enough to capture the maximum impact even if the maximum does not occur for several thousand years. For the inadvertent intruder analysis, the sensitivity and uncertainty analysis is limited to qualitative arguments. These analyses, together with the sensitivity and uncertainty analyses discussed below, were used in this draft 3116 Determination and meet the intent of the 10,000-year performance period.

The NRC's LLW PA guidance suggests a two-part approach to establishing the time of analysis (NRC 2000). The first part is a 10,000-year analysis period and the second part is a qualitative evaluation of the analysis beyond 10,000 years to identify any significant deficiencies in disposal system

performance. In the TFF PA, groundwater contaminant transport simulations (i.e., the dominant dose pathway) are conducted for three radionuclides (^{90}Sr, ^{99}Tc, and ^{129}I) that provide the majority of the groundwater pathway doses during the 1,000-year analysis period.

The PA's performance period of 1,000 years after INTEC TFF closure includes an institutional control period and a post-institutional control period, while the sensitivity analysis for the groundwater pathway provides simulations out past 10,000 years. The use of the groundwater modeling past 10,000 years, coupled with the intruder scenarios, which peak immediately after the institutional control period (100 years following TFF closure), provides for compliance with the 10,000-year period of performance as suggested by NRC's guidance.[hh]

The PA assumes an institutional control period at the INL Site of 100 years following TFF closure. Developing an assumption for institutional controls in performance assessments includes consideration of site-specific land use planning and planning for engineered controls such as active barriers (e.g., caps, fences, and security) and passive barriers (e.g., signs, monuments, and markers). The *INEEL Comprehensive Facility and Land Use Plan* (DOE-ID 1997) describes government control of the site for at least the next 100 years. The INL has not prepared specific plans or designs for active or passive controls following this timeframe. Consistent with guidance in NRC's LLW PA (NRC 2000), a 100-year post-closure institutional control period has been selected.

During this institutional control period, periodic maintenance and monitoring activities are assumed to be conducted, and active controls such as guards and fences are expected to be used. During the institutional control period, members of the public are assumed to reside at the INL Site boundary and intruders are assumed to be restricted from entering the site (DOE-ID 1995). During the post-institutional control period (2112–3012), the INL Site will not be maintaining the facility, so the facility could be accessed by the public. During this period, members of the public are assumed to reside as close as 100 m (328 ft) from the TFF. The PA evaluates the degradation of the various barriers (e.g., grout, tanks, vaults, and piping) to radionuclide release. Although models predict that these barriers will last 40,000 years, the TFF PA assumes a tank and tank grout life of only 500 years (NRC 2000), and a life of only 100 years for the vaults and grout system between the tank and the vault outer walls. The groundwater model analysis shows that the contamination plume center (where the highest concentrations enter the regional aquifer) would be 600 m (1,969 ft) southward in the downgradient direction from the center of the southernmost TFF tank. The contamination plume center is taken as the source of drinking water after the institutional control period of 100 years has expired. This center is the distance from the TFF for the highest calculated radiation dose. The intruder scenario considers contamination spread from two activities: a 3-m (10-ft) excavation for a basement and a well drilled through the top of the tank. The maximum dose would occur immediately after institutional control ended at 100 years after closure (2112).

The dose posed by the residual waste inventory at closure, the general requirements for disposal in a near surface facility described in 10 CFR 61.40, and the performance objectives for LLW disposal described in 10 CFR 61.41–61.44 are discussed in the following subsections.

7.1 Analysis of Performance

The conceptual models developed and the computational approach used to assess the performance of the TFF are also described fully in the PA. The conceptual models are derived from technical

hh. The contaminant transport simulations are evaluated out to 1 million years for these radionuclides. Prior modeling simulations for the TFF assessed the remaining radionuclides for a period of 1 million years. The results of this prior modeling indicate that the remaining radionuclides are not of concern for the TFF PA (DOE-ID 2003b).

information presented in Section 2 of the PA (DOE-ID 2003b). These models embody a number of simplifying assumptions to facilitate the computational analysis required to assess the long-term performance of the TFF. The dose calculations are presented in Subsection 3.3.5 of the PA.

7.1.1 Institutional Control Period

The institutional control period is the 100-year time interval specified in the *INEEL Comprehensive Facility and Land Use Plan* (DOE-ID 1997) following closure of the TFF. The assumed closure date of the TFF is currently 2012. Periodic maintenance and monitoring activities are conducted during the institutional control period. The TFF site is assumed to be stabilized and no longer operational during this period but will remain part of the INL, and therefore, fenced and patrolled to eliminate the possibility of inadvertent intruders.

7.1.2 Post-Institutional Control Period

The final time period of concern is when the facility is no longer maintained by the INL Site and could be accessed by the public. The total duration of this period is 1,000 years from the time of closure; however, to determine groundwater dose, which may peak at a later time, simulations were carried out to 1 million years. Therefore, the 10,000-year period of performance is addressed by the PA. Projections of conditions and activities after 1,000 years are uncertain and difficult to assess. In addition, peak doses may not occur from TFF releases until several thousand years after closure because of the presence of long-lived radionuclides. The potential peak doses from these long-lived radionuclides are in the sensitivity analysis.

7.1.3 Exposure Pathways

Numerous exposure pathways were analyzed in the TFF PA to demonstrate conformance with a variety of performance objectives. These exposure pathways include: (1) drinking water dose from groundwater,[ii] (2) all-pathways dose from groundwater, and (3) air dispersion pathways.[jj] In addition, the intruder pathway and radon flux analyses were also considered and are described in Subsections 3.3.4 and 5 of the TFF PA (DOE-ID 2003b). Only the all-pathways dose assessment and intruder analysis are applicable to demonstrate compliance with the NDAA criteria and will be further discussed in this draft 3116 Determination. Intruder scenarios are discussed in Subsection 7.2.

The TFF PA includes a comprehensive all-pathways exposure scenario. The primary mechanism for transport of radionuclides from the TFF is expected to be leaching to the groundwater and subsequent

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ii. The groundwater protection standard chosen for evaluation in the PA is a drinking water dose of 4 mrem/yr. The nearest location from the disposal site for off-Site members of the public depends on the time period after disposal. During the period of active institutional control for the first 100 years after facility closure, off-Site members of the public are assumed to be located no closer to the disposal site than the present boundary of the INL. However, after active institutional control ceases, off-Site members of the public could be located as close as 100 m (328 ft) from the TFF. However, for the groundwater pathway, the member of the public is assumed to have a well located in the groundwater at the point of maximum concentration, which for the analyses presented in the PA is located 600 m (1,969 ft) from the TFF.

The primary mechanism for transport of radionuclides from the TFF is expected to be leaching of radionuclides to the groundwater and subsequent human consumption. Thus, in the dose analysis for groundwater protection, an off-Site member of the public is assumed to use water from a well for domestic purposes. The well is assumed to be located where the maximum concentrations of radionuclides in groundwater are predicted to occur.

jj. Volatile radionuclides ^{3}H and ^{14}C were evaluated in the air dispersion scenario. The depth to the waste and the waste physical characteristics (i.e., grouted waste) limits the analysis to volatile radionuclides in the waste form. Additional details on the air dispersion scenario are provided in Subsection 3.3.5 of the PA (DOE-ID 2003b).

human consumption. Thus, in the all-pathways dose analysis, an off-Site member of the public is assumed to use water from a well for domestic purposes. The well is assumed to be located where the maximum concentrations of radionuclides in groundwater are predicted to occur.

The following exposure pathways involving the use of contaminated well water are assumed to occur:

- Direct ingestion of contaminated water
- Ingestion of milk and meat from dairy and beef cattle that drink contaminated water
- Ingestion of vegetables grown in garden soil irrigated with contaminated water
- Ingestion of milk and meat from dairy and beef cattle that eat fodder from pasture irrigated with contaminated water.

Additional exposure pathways for off-Site members of the public could involve releases of radionuclides into the air (i.e., volatile radionuclides). Exposures from the air pathway also are considered in the PA.

7.2 Models and Assumptions

7.2.1 Source Term Model

For the TFF concrete vaults, release rates of radionuclides were estimated using the Disposal Unit Source Term-Multiple Species (DUST-MS) computer code (Sullivan 1993, 2001). One-dimensional DUST-MS transport simulations were conducted for radionuclide sources in the grouted tank, piping, and the sandpad beneath the tank.

7.2.1.1 DUST-MS Model Description. DUST-MS considers four major processes: fluid flow, container degradation, waste form leaching, and contaminant transport. The DUST-MS model permits selection of a unique failure time and waste form type for each container. To simulate different waste forms, DUST-MS has four different models to estimate release rates: rinse with partitioning, diffusion, uniform degradation, and solubility-limited release. For these four processes, DUST-MS computes release rates with an analytical model or a finite difference model. After calculating waste form releases, the movement of the contaminants through subsurface system materials is determined using a one-dimensional finite difference procedure with material- and contaminant-specific distribution coefficients, diffusion coefficients, dispersion coefficients, and bulk densities. The governing transport equation simulates the distribution and movement of contaminants from advection, dispersion, diffusion, radioactive production and decay, sorption, and sources and sinks external to the containers. The DUST-MS code has received extensive testing and verification. The DUST-MS code predictions compare favorably to known analytical solutions as well as other code predictions (DOE-ID 2003b).

7.2.1.2 Conceptual Model. The conceptual model for computing release rates of radionuclides out of the TFF vault is illustrated in Figure 17. Infiltrating water from the ground surface contacts the waste inside the vaults.

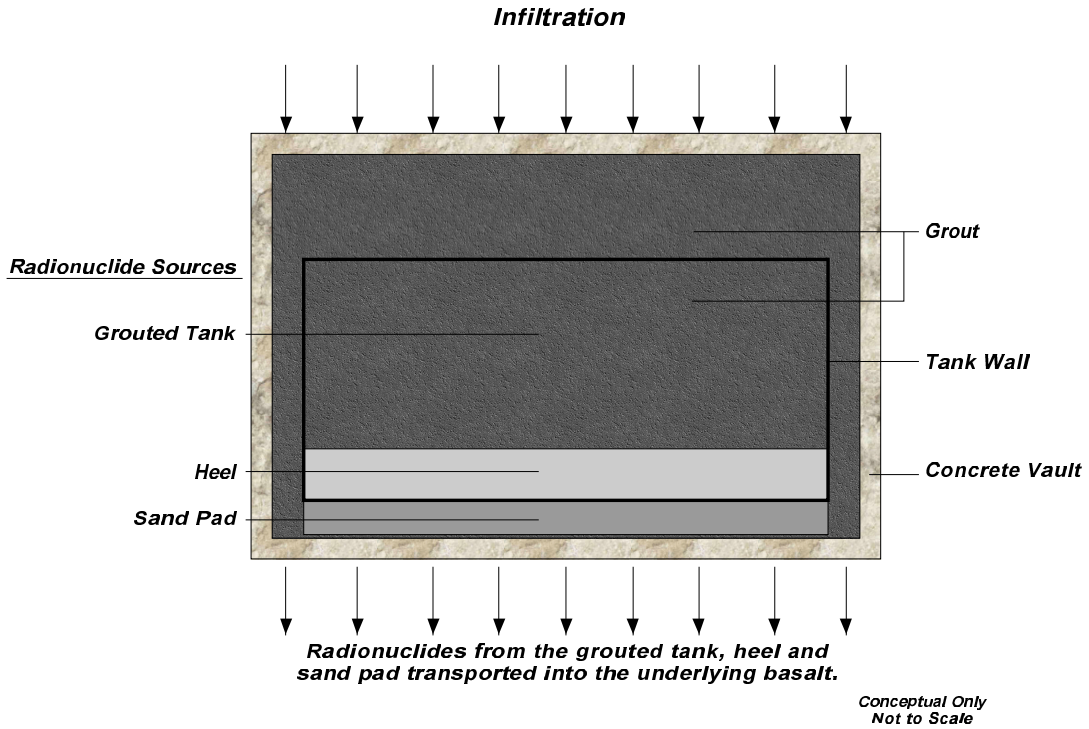

Figure 17. The DUST-MS conceptual model for release from the TFF vaults and tanks (DOE-ID 2003b).

In Figure 17, radionuclides are assumed to be located in the grouted tank heel and in the sandpad beneath the tank. The release of radionuclides from the grouted tank heel and from the sandpad is modeled in DUST-MS assuming surface rinsing. The vault is assumed to remain intact for 100 years. The tanks are assumed to remain intact for 500 years. Upon vault and tank failure, radionuclides are released from the two sources.

The DUST-MS code calculates release rates out of the vault and tanks into the unsaturated zone. Release rates are computed for transport through the grout inside the tank (15 cm [6 in.]), the sandpad (15 cm [6 in.]), and the degraded concrete (0.76 m [2.5 ft]). Retardation is assumed to occur in the waste release model (surface rinsing), in the grout inside the tank, in the sandpad, and in the degraded concrete.

7.2.2 Groundwater Flow and Transport Modeling

A two-dimensional unsaturated/saturated PORFLOW model was used to simulate water and contaminant transport in the subsurface at the INTEC facility (see Figure 18) (ACRi 2000). This approach allowed a detailed approximation of the complex geology underlying the facility. The two-dimension problem domain represented a vertical slice in a north-south direction beginning at the Big Lost River, through the center of two tank vaults, and southward in the downgradient direction for a total distance of 2,500 m (8,202 ft). The top of the model is located at land surface and extends to a depth of 200 m (656 ft)—well below the top of the water table located at 134–139 m (440–456 ft) below ground surface. By default, the unit thickness of the two-dimensional model in an east-west direction is 1 m (3 ft).

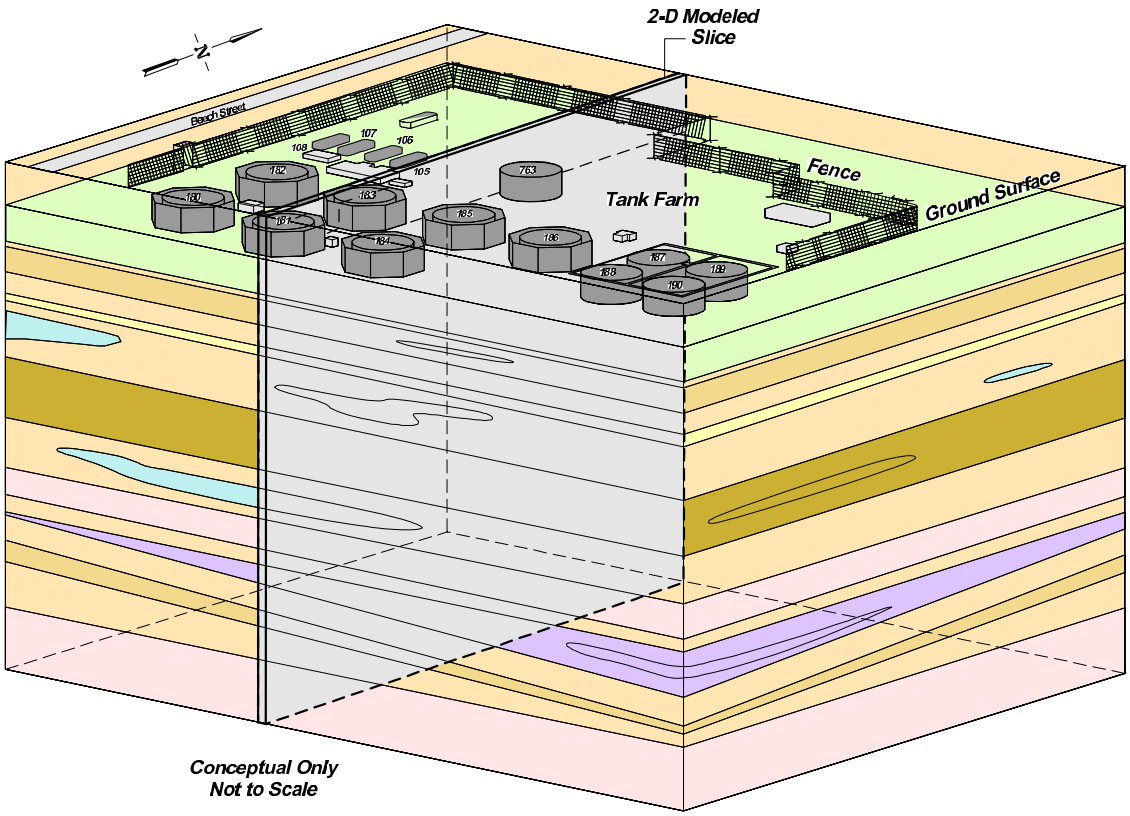

Figure 18. Illustration of the two-dimensional modeling slice used in PORFLOW (DOE-ID 2003b).

As noted earlier, DUST-MS was used to predict the release rates from the various waste forms. Radionuclide releases coincided with degradation rates of the waste forms discussed in Subsection 3.1.2.2 of the PA (DOE-ID 2003b). Release rates calculated by DUST-MS were incorporated as input in PORFLOW, with the initial radionuclide releases beginning at the time of degradation. For example, the outer vault began to release radionuclides after 100 years and the tanks/piping after 500 years. These assumptions are conservative, based on the best estimates provided from the degradation analysis.

An important component of the conceptual model for the transport of contaminants is the relationship of dispersion in the two-dimensional model. Since the model uses a unit thickness of 1 m (3 ft), there is no lateral dispersion of contaminants beyond this thickness. It is assumed that contaminants disperse or diffuse out of the lateral boundary at the same rate as contaminants move into the unit thickness of the model domain. Essentially, all dispersion of contaminants occurs in the longitudinal direction, with zero transverse dispersion. This is a conservative approach to predicting downgradient contaminant concentrations.

Since the model domain is located in the center of the tanks, the highest contaminant concentrations are located in the source area of the model. Although it is assumed that the same amount of contaminants disperse in a transverse direction out of, as into the model domain, in reality there would be the same loss of contaminants, albeit small, in the transverse direction. Consequently, downgradient contaminant concentrations will be slightly higher for the two-dimensional simulation compared to a fully three-dimensional simulation.

Radionuclide concentrations were observed downgradient in the regional aquifer at the location where the highest concentrations occur. During unsaturated flow simulations, it was observed that the perched water zones deflected the contamination. Consequently, the usual 100-m (328-ft) downgradient location was inappropriate to quantify the maximum impacts of radionuclide releases from the tanks and vaults. Based on the radionuclide concentrations distribution, the maximum observed concentrations occur approximately 600 m (1,969 ft) downgradient from the tanks.

The intruder scenarios were screened based on the physical aspects of the facility (i.e., depth to the waste). Intruder scenarios considered for use in the PA were those previously described for low-level radioactive waste PAs (Kennedy and Peloquin 1988; NRC 1982). These intruder scenarios include both acute and chronic exposure scenarios. Acute exposure scenarios involve exposures of short duration, and include an intruder-construction scenario, a discovery scenario, and a drilling scenario.

7.2.2.1 Intruder-Construction Scenario. The intruder-construction scenario involves an inadvertent intruder who chooses to excavate or construct a building on the disposal site. In this scenario, the intruder is assumed to dig a basement excavation to a depth of approximately 3 m (10 ft) (Oztunali and Roles 1986). It is assumed that the intruder does not recognize the hazardous nature of the material excavated. He or she is exposed to radioactive constituents in the waste during the excavation of the basement. The intruder also is exposed to the exhumed waste by inhalation of resuspended contaminated soil and external irradiation from contaminated soil.

The depth to the residual waste at the bottom of the tanks and/or sandpads (i.e., 3 m [10 ft] of soil plus 9 m [30 ft] of concrete) would preclude direct contact with the waste from the 3-m (10-ft) excavation. However, approximately 30% of the contaminated piping associated with the tanks is located within 3 m (10 ft) of the surface.

Due to the disposal depth of the waste in the tanks at the TFF (i.e., greater than 10 m [33 ft]) and the depth of the sandpad contamination, the intruder-construction scenario was not considered applicable to these waste. However, the intruder-construction scenario was considered for the radionuclide inventory located in the associated piping less than 3 m (10 ft) from the surface.

7.2.2.2 Intruder-Drilling Scenario. The intruder-drilling scenario assumes the short-term exposure of a hypothetical intruder to drill cuttings from a borehole penetrating the waste disposal site. This scenario involves wastes buried below the depth of typical construction excavations.

Oztunali and Roles (1986) indicate that for waste below 10 m (33 ft), the only applicable intrusion scenario is the intruder-drilling scenario. They also note that for grouted waste or waste disposed of in reinforced concrete structures, a time period of 500 years after site closure is assumed as the effectiveness limit for this waste form. Therefore, this scenario is not considered applicable for drilling through reinforced concrete until 500 years post-closure. However, they also note that the scenario is assumed to be fully applicable for grouted waste at any time. Because the waste could be contacted and moved to the surface, thereby exposing the intruder, the intruder-drilling scenario was retained for analysis in the TFF PA (DOE-ID 2003b).

7.2.3 Chronic Intruder Exposure Scenarios

These scenarios are described below. Those scenarios not considered applicable to the TFF were screened out from further consideration.

7.2.3.1 Intruder Post-Construction Scenario. The chronic intruder post-construction (i.e., agriculture) scenario is an extension of the acute intruder-construction scenario. It is assumed in this scenario that an intruder lives in the building constructed as part of the intruder-construction scenario and engages in agricultural activities on the contaminated site. The intruder is exposed to contamination by inhalation of resuspended contaminated soil, inhalation of gaseous radionuclides released from the waste, external irradiation, ingestion of contaminated soil, ingestion of contaminated beef and milk, and ingestion of contaminated vegetables.

As stated earlier, the intruder-construction scenario was only considered applicable to the piping associated with the TFF that is less than 3 m (10 ft) from the surface. The depth of the residual waste in the tanks and the sandpads prevents its excavation to the surface. Therefore, the intruder-agriculture scenario was retained for the piping inventory.

7.2.3.2 Post-Drilling Scenario. The chronic post-drilling scenario is an extension of the acute drilling scenario. It assumes that the intruder occupies the site after drilling a water well and grows crops on a mixture of clean soil and contaminated drill cuttings. After exhumation of the waste, the exposure pathways are the same as for the intruder-agriculture scenario. This intruder scenario was retained for further analysis.

7.3 Acute Intruder-Drilling Scenario Definition

The acute intruder-drilling scenario assumes that an inadvertent intruder drills a well into the contents of the tank and vault system. The intruder is exposed to contaminated drill cuttings spread over the ground and contaminated airborne dust. In the standard drilling scenario used in many PAs, the intruder is assumed to be exposed to contaminated drill cuttings in a mud pit. However, site-specific information developed through interviews with local well drilling contractors in the Idaho Falls area indicates that drillers spread the cuttings over the ground and do not use mud pits (Seitz 1991). The authors of the Radioactive Waste Management Complex PA (Maheras et al. 1997) used this site-specific deviation of the standard drilling scenario; it also was incorporated into the TFF intruder-drilling scenario. The assumption that the drill cuttings are spread over the ground will result in higher dose estimates than if the cuttings were assumed to be in a mud pit because of the decrease in the shielding factor.

The drill cuttings are assumed to be spread over a 2,200-m^2 (23,681-ft^2) lot, which corresponds to about 0.5 acre. Typical lot sizes located outside of the Idaho Falls city limits are typically 1–3 acres. Therefore, a 2,200-m^2 (23,681-ft^2) lot size was considered conservative for use in the Radioactive Waste Management Complex PA (Maheras et al. 1997), and also was incorporated into the TFF intruder-drilling scenario.

Well drilling contractors in the Idaho Falls area have reported that two types of wells are typically drilled: small-diameter residential wells and large irrigation wells. The small residential wells are typically 15–20 cm (6–8 in.) in diameter, serve a single residence, and may provide enough water for a family garden and small quantities of livestock. The large-diameter irrigation wells are drilled to serve systems that irrigate hundreds of acres; the wells are located in the middle of farm fields, not near the farmer's residence. Therefore, a farmer would not drill an irrigation well to acquire water for his residence. Large-diameter irrigation wells are currently drilled in 46-cm (18-in.) diameters but drilling contractors indicated that 56-cm (22-in.) diameter irrigation wells would be drilled in the future (Seitz 1991). An acute drilling exposure could result from either drilling a 15-cm (6-in.) diameter residential well or a 56-cm (22-in.) diameter irrigation well. The larger 56-cm (22-in.) diameter irrigation well was assessed for the acute intruder-drilling scenario.

The intruder is assumed to reside by the contaminated cuttings for 160 hours, the time local Idaho Falls well drilling contractors say it would take to drill and develop a 56-cm (22-in.) diameter irrigation well (Seitz 1991). The exposure pathways for this acute drilling scenario include inhalation of resuspended drill cuttings, external exposure to the ground source, and inadvertent soil ingestion. Figure 19 illustrates the acute intruder-drilling scenario.

The TFF concrete vaults are reinforced concrete but the grout between the tank and vault and also inside of the tank is not reinforced concrete. However, considering that the depth to the waste in the tank is 10 m (33 ft), that the waste is contained within a stainless steel tank, and that the overlying concrete vault roof is reinforced, credit beyond 100 years could be taken for a barrier to intrusion. However, the intruder-drilling scenario was assumed to begin 100 years after closure.

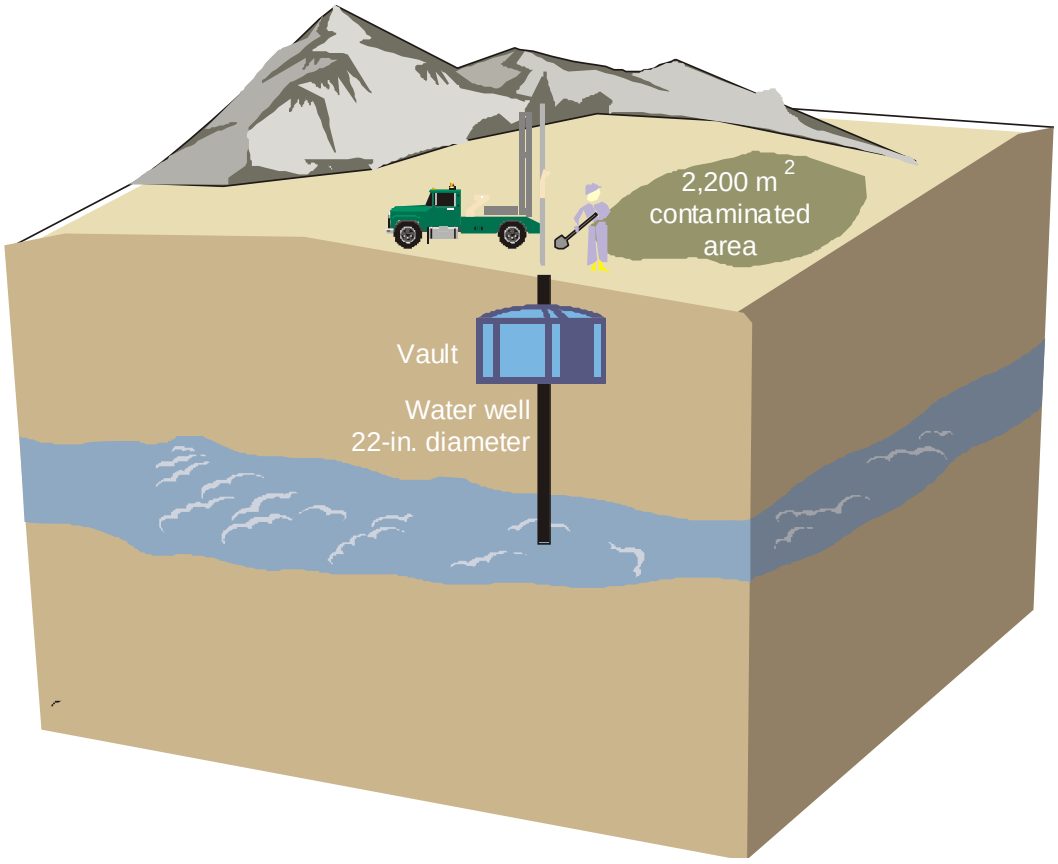

Figure 19. Graphical representation of the acute intruder-drilling scenario (DOE-ID 2003b).

7.4 Acute Intruder-Construction Scenario

The acute intruder-construction scenario assumes that an inadvertent intruder moves onto the TFF and excavates a 20- by 10- by 3-m (66- by 33- by 10-ft) excavation for a basement (Oztunali and Roles 1986).

The intruder is assumed to spend 160 hours working on the excavation of the basement. The exposure pathways for this acute scenario include inhalation of resuspended soil, external exposure to the ground surface, and inadvertent soil ingestion.

The determination of the unit concentration dose factors for each exposure pathway was determined using the equations in Subsection 5.2 of the PA (DOE-ID 2003b) for the acute intruder-drilling scenario with the exception that the external dose factors for contaminated soil were based on infinite thickness and infinite lateral extent from Federal Guidance Report 12 (EPA 1993).

The activity concentration of radionuclides in the excavation were determined by assuming that the radionuclide inventory in the 30% of the piping located within 3 m (10 ft) of the surface was located over the 5-acre (20,234-m^2 [217,797-ft^2]) area of the TFF.

The results of the intruder-construction analyses are given in Subsection 5.6 of the PA (DOE-ID 2003b).

7.5 Chronic Intruder Post-Drilling Scenario Definition

The chronic intruder post-drilling scenario assumes that an inadvertent intruder moves onto the TFF and drills a residential well into the waste. The drilling portion of the scenario evaluates a 15-cm (6-in.) diameter residential well. This type of well serves a single residence and provides sufficient water for a family garden and small quantities of livestock. As described in the acute intruder-drilling scenario, large-diameter wells are drilled to serve irrigation systems (i.e., hundreds of acres) that are located in the middle of farm fields, not near a farmer's residence. Therefore, in the chronic post-drilling scenario, the residence/home garden is evaluated using the traditional drinking water well diameter of 15 cm (6 in.).

The drill cuttings that are brought to the surface are assumed to be spread over 2,200 m^2 (23,681-ft^2) (approximately 0.5 acre) of land surface. The waste is assumed to be mixed to a depth of 0.6 m (2 ft). The mixing depth of 0.6 m (2 ft) is based on using a deep tilling plow to increase the depth of the root zone and to break up soil compaction. To minimize erosion these plows are used in areas of southeastern Idaho with highly erodible soils. Deep tilling plows have shanks that till to a depth of 0.6 m (2 ft) and are sold at Idaho Falls implement dealers (Maheras et al. 1997).

The chronic post-drilling scenario assumes that the intruder is exposed to the drill cuttings during plowing and cultivation (i.e., dust inhalation). In addition, the intruder is assumed to ingest contaminated food products from the garden and from beef and milk cattle consuming contaminated forage. The intake of contaminated forage by cattle was adjusted according to the fraction of feed grown on contaminated cuttings and the necessary remaining feed obtained from uncontaminated ground. Figure 20 illustrates the chronic intruder post-drilling scenario.

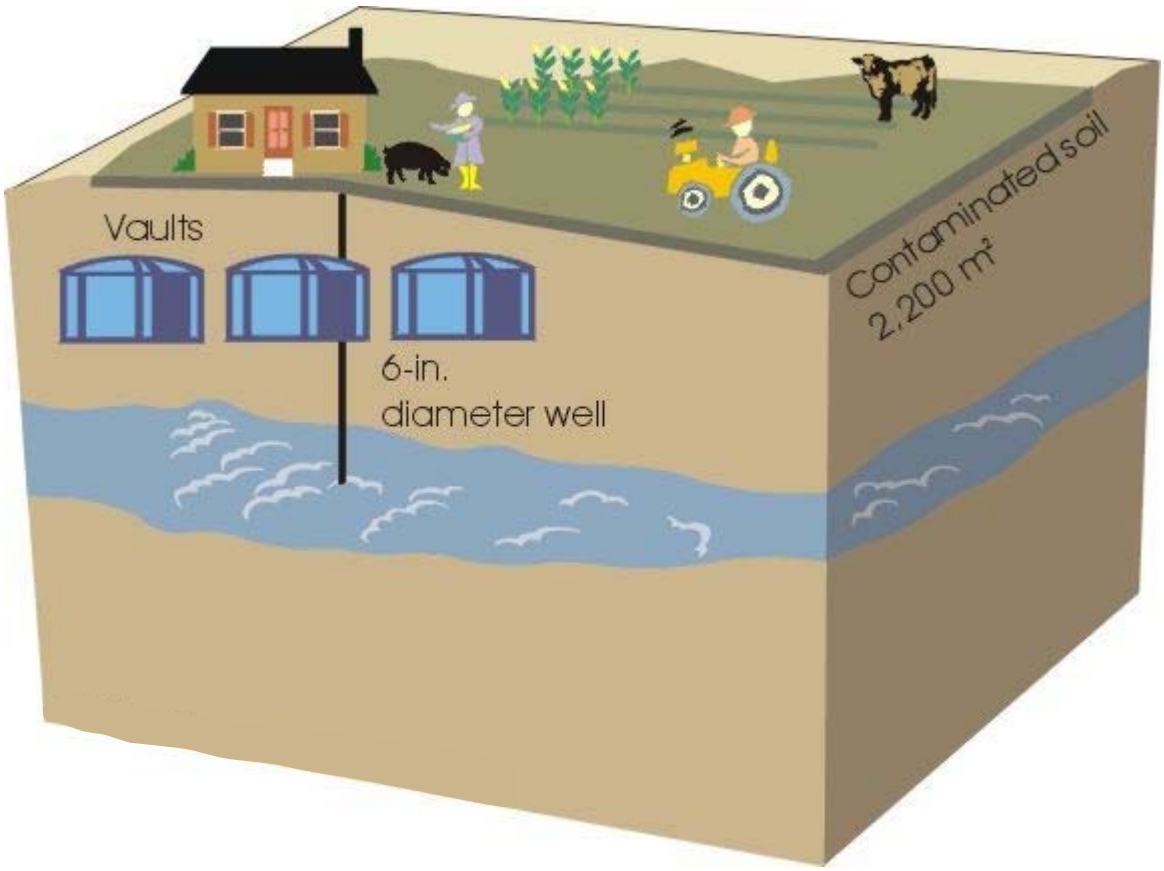

Figure 20. Chronic intruder post-drilling scenario (DOE-ID 2003b).

7.6 Chronic Intruder Post-Construction Scenario Definition

The chronic intruder post-construction scenario assumes that an inadvertent intruder moves onto the TFF, excavates a basement, spreads the excavated material over the land, and subsequently farms the area.

The excavation soil is assumed to be spread over 2,200 m^2 (23,681 ft^2) (approximately 0.5 acre) of land surface. The waste is assumed to be mixed to a depth of 0.6 m (2 ft), which is based on using a deep tilling plow to increase the depth of the root zone and to break up soil compaction. These plows are used in areas of southeastern Idaho with highly erodible soils to minimize erosion. Deep tilling plows have shanks that till to a depth of 0.6 m (2 ft) and are sold at Idaho Falls implement dealers (Maheras et al. 1997).

The chronic intruder post-construction scenario assumes that the intruder is exposed to the excavated soil during plowing and cultivation (i.e., dust inhalation). In addition, the intruder is assumed to ingest contaminated food products from the garden and from beef and milk cattle that consume contaminated forage. The intake of contaminated forage by cattle was adjusted according to the fraction of feed grown on contaminated cuttings and the necessary remaining feed obtained from uncontaminated ground.

7.7 NRC Review, and Recommendations for Future Work

The NRC reviewed the PA in 2002 in consultation with DOE. Table 17 presents the conclusions reached by the NRC and present recommendations for future work. Since 2002, DOE has addressed the recommendations and has prepared analysis to support the PA. Each criterion reviewed by NRC is shown followed by NRC's conclusions and then by recommendations and actions taken by DOE.

Table 17. Summary of NRC review conclusions and recommendations (SECY-93-0079, 2003).

<table>
<tr><td colspan="2">Criterion: The waste should be processed (or should be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical.</td></tr>
<tr><td colspan="2">Conclusions:
The U.S. Department of Energy Idaho Operations Office (DOE Idaho) methodology for solid mass estimation and liquid volume estimation is technically adequate
The conservative source term is likely to bound the residual materials concentrations and quantities actually remaining in the tanks
The DOE Idaho's argument that key radionuclides will be removed to the extent technically and economically practical is reasonable.</td></tr>
<tr><td>Recommendations</td><td>DOE Actions</td></tr>
<tr><td>Sampling of the radiological composition of residual materials remaining in tanks after cleaning should be completed before tank grouting and final closure, in accordance with DOE Idaho's SAP.</td><td>Characterization of the tank contents is updated as tanks are cleaned. Tanks WM-180 through WM-186 and WM-103 through WM-106 have been cleaned and sample data have been collected since cleaning began in 2002. The residual waste inventory at closure is significantly lower than estimated in the PA.[a]</td></tr>
<tr><td>Because of the cooperative physical characteristics of the residual materials remaining in the tanks and the relatively small economic impact associated with tank flushing, DOE-Idaho should follow its current plan for cessation of tank flushing only after removal of residual activity from the tank becomes insignificant.</td><td>DOE Idaho has followed the recommendation during tank cleaning that tank flushing only be stopped after removal of residual activity from the tanks becomes insignificant. The post-cleaning videos and samples indicate that only a very small amount of residual remains after the cleaning process is completed.</td></tr>
<tr><td>DOE Idaho should stay abreast of tank cleaning technology for potential use in future tank cleaning, if such technology is technically and economically practical.</td><td>DOE Idaho continues to stay abreast of tank cleaning technology. However, as noted in this report, tank cleaning has been very successful with the residual inventories being less than the best estimates presented in the PA.[a]</td></tr>
</table>

Table 17. (continued).

<table>
<tr><td colspan="2">Criterion: The waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR 61, Subpart C are satisfied.</td></tr>
<tr><td colspan="2">Conclusions:
• As indicated by the DOE Idaho PA, combined doses to the public form all-pathways are projected to be well below the 25-mrem/yr limit; therefore, staff considers that there is reasonable assurance that safety requirements comparable to 10 CFR 61.41 can be satisfied, including ALARA requirements
• Staff considers that there is reasonable assurance that safety requirements comparable to 10 CFR 61.42 for protection of individuals from inadvertent intrusion can be satisfied
• The worker is protected by DOE regulations that are comparable to 10 CFR 20; therefore, the worker protection performance objective (10 CFR 61.43) can be considered to be met
• The DOE Idaho's plans to fill the tanks, vaults, and ancillary piping with multiple layers of reducing grout appear sufficient to indicate that safety requirements comparable to 10 CFR 61.44 can be met.</td></tr>
<tr><td>Recommendations</td><td>DOE Actions</td></tr>
<tr><td>If sampling after tank cleaning indicates that the source term is significantly larger than that used in the current PA, then the PA should be reevaluated.</td><td>The results of the tank samples after tank cleaning have been used by DOE Idaho to ensure that the source term assumed in the PA is not exceeded. The results of these analyses are documented in engineering design files.[b] To date, all tank samples after cleaning indicate that the residual waste inventory at closure is much less than that assumed in the PA. A methodology to calculate radiation dose based on inventories of cleaned tanks has been developed.[c]</td></tr>
<tr><td>Although this assessment assumed that the conservative sorption coefficients for concrete, basalt, and interbedded sediments were sufficiently bounding, DOE Idaho should consider expanding its literature review or conducting laboratory testing to provide additional confidence for the assertion of conservatism. Currently, the conservative values are simply calculated by interpolation between a lower bound and a realistic case. If retardation of ^{99}Tc in the degraded concrete layer at the base of the tanks provides a significant performance effect, a technical basis should be established for the assumption of reducing conditions in that location.</td><td>DOE Idaho has completed an additional sorption coefficient report[d] that compares the sorption coefficients used in the PA against values published in the literature. This additional literature review indicated that the sorption coefficients used in the PA appear to be reasonable.[a]

Additional sensitivity analysis has been performed to determine the effect of the vertical location of residual waste in the stabilized tank.[e]

As explained in the errata to the PA, the description of the ^{99}Tc vault sorption coefficient was incorrect. In fact, an oxidizing sorption coefficient was used for the vault concrete, while a reducing sorption coefficient was assumed for the grouted waste. Additional, analysis has been performed to understand the transport time to the aquifer and the resulting dose from ^{99}Tc if oxidizing conditions are assumed for the grouted waste in the tanks.</td></tr>
</table>

Table 17. (continued).

Recommendations	DOE Actions
Future PA analyses should evaluate the sensitivity of the results to the use of oxidizing condition distribution coefficients for grout.	Additional analysis has been performed to understand the transport time to the aquifer and the resulting dose from ^{99}Tc if oxidizing conditions are assumed for the grouted waste in the tanks. The results of this analysis indicate the ^{99}Tc tank peak drinking water dose occurs at 842 years at a dose of 0.54 mrem/yr. The total drinking water dose peaks at 874 years due to the combined doses from ^{99}Tc and ^{129}I. The drinking water dose contributions from ^{99}Tc and ^{129}I to the total drinking water dose at this time are 0.54 and 0.76 mrem/yr, respectively.[f]
DOE Idaho should evaluate, and if needed, enhance QA controls of documentation in future PAs as the TFF closure progresses.	DOE is committed to QA. To enhance QA, more thorough internal and independent reviews have been conducted on the PA and associated documentation. Internal reviews have been performed on this draft 3116 Determination.
As cleaning and closure of tanks progress, the closure strategy for each tank should be refined based on information obtained from prior tank and ancillary equipment closures at the TFF.	The closure strategy for each tank has continually been updated from lessons learned from prior tank cleanings and ancillary equipment closures. The cleaning and closure method has been refined such that efficiencies in removal of the residual tank inventories are being obtained.
DOE Idaho should investigate methods for measuring or better estimating the contaminated sandpad radionuclide inventories.	DOE has considered alternative options to collect sandpad samples; however, no sampling method either by direct or indirect means has been found, which is either practical or would provide data of known quality. DOE has evaluated options such as flushing the vault and collecting samples of flush water. It has been concluded that this option would not provide an improved estimate of sandpad source term. Contamination from the sandpad is not distinguishable from other sources of contamination from the vaults. DOE has developed a bounding inventory for the sandpad in the PA. The sandpads are located beneath the tanks, making sampling difficult to impossible. The sandpad inventory is considered by DOE Idaho to bound the real contamination potential. Several vault flooding events from melting snow have not been considered in the final sandpad inventory calculations, which would reduce the sandpad inventory.

a. DOE-ID 2003b.
b. Portage 2005a, 2005b, 2005c, 2005d, 2005e, 2005f, 2005g.
c. Portage 2005i.
d. Portage 2005j.
e. Portage 2005k.
f. Portage 2005l.

7.8 Estimated Radiation Dose Using the Current Estimate of the Residual Waste Inventory at Closure

Performance objectives for radioactive waste land disposal are provided in 10 CFR 61, Subpart C. Table 18 compares the dose based on the conservative residual waste inventory used in the PA and the dose posed by the current estimate of the residual waste inventory at closure (described in Section 2) with the listed performance objectives. Results of the PA analyses indicate compliance with the performance objectives shown in Table 18. The dose for the current estimate of the residual waste inventory at closure for the TFF tank system is less than the dose calculated for the inventory used in the PA. The doses by radionuclide for the groundwater all-pathways, acute intruder-drilling scenario, and chronic intruder-drilling scenario are less than the doses predicted in the PA. The dose comparison shows a decline in dose to various receptors.

Table 18. Comparison of performance assessment results to predicted dose at closure (DOE-ID 2003b).

Performance Objectives	PA Results	Current Estimate of Residual at Closure
All-pathways dose to the public (Not exceeding 25 mrem/yr)	1.86 mrem/yr	0.46 mrem/yr
Acute-drilling scenario (less than 500 mrem)	232 mrem	152 mrem
Acute-construction scenario (less than 500 mrem)	0.80 mrem	0.23 mrem
Chronic post-drilling scenario (less than 500 mrem/yr)	91.1 mrem/yr	25 mrem/yr
Chronic post-construction scenario (less than 500 mrem/yr)	26.1 mrem/yr	3.15 mrem/yr

Notes:

1. The groundwater pathway contributed 1.35 mrem/yr.

2. The peak annual dose to the thyroid is approximately 6 mrem/yr compared to the 10 CFR 61.41 limit of 75 mrem/yr.

3. The peak annual dose to any other organ is approximately 0.15 mrem/yr compared to the 10 CFR 61.41 limit of 25 mrem/yr.

For the residual waste inventory at closure, the Tanks WM-180 and WM-182 inventories are used as a basis for the estimated doses shown in the last column of Table 18. Tanks WM-180 and WM-182 are used because this tank combination has the largest residual waste inventory of the cleaned 300,000-gal tanks. The modeling in the PA was designed to use the inventory from a set of tanks that are oriented north to south to match the regional groundwater flow. The radiation doses are estimated using a dose conversion factor (DCF), which is the ratio between the radiological inventory (in Ci) and the radiological dose (in mrem/yr), as determined in the PA. Although the DCF is a ratio determined from data used in the PA, the same ratio is applicable to the residual material present in a cleaned tank at the completion of closure activities. Therefore, since the post-closure inventory of the cleaned tanks is known, the exposure scenario doses may be determined using the DCF. A description of the methodology and results are shown in "Radiological Dose Calculations for the Tank Farm Facility at the Idaho Nuclear Technology and Engineering Center" (Portage 2005m). The ratio for DCFs is shown as an equation below.

$$\frac{PA_d}{PA_i} = DCF$$

$$DCFxCR_i = CR_d$$

where:

PA_i = performance assessment inventory

PA_d = performance assessment radiation dose

DCF = dose conversion factor

CR_i = current residual inventory

CR_d = current inventory radiation dose.

Since the PA inventory, the PA radiation dose, and the current estimate of residual inventory are known, the current residual inventory radiation dose can be calculated using the simple ratio shown above.

7.9 10 CFR 61.40, "General Requirement"

The general requirement of 10 CFR 61.40 states:

> Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in §§ 61.41 through 61.44.

The TFF PA was developed under similar guidance and requirements from DOE, which required the analysis to provide a "reasonable expectation" that the performance objectives are not exceeded as a result of operation and closure of the facility. The PA meets the performance objective of "reasonable assurance" by using conservative estimates of parameter values, concrete degradation rates, and human activity models. In addition, a sensitivity and uncertainty analysis is used to ensure that a conservative analysis was conducted for the TFF closure that bounds the potential final closure status of the facility.

The following subsections compare each of the 10 CFR 61 performance objectives with those used in the TFF PA analyses.[kk]

7.10 10 CFR 61.41, "Protection of the General Population from Releases of Radioactivity"

The measure for protecting the public from radioactive material that may be released from the disposal facility by any pathway is established in 10 CFR 61.41. The NRC requirement states:

> Concentrations of radioactive material which may be released to the general environment in groundwater, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 mrems to the whole body, 75 mrems to the thyroid, and 25 mrems to any other organ of any member of the public. Reasonable effort should be

kk. The NRC issued the requirement for LLW disposal, 10 CFR 61, on December 27, 1982. The requirement was developed to provide specific regulations for the disposal of LLW in near-surface disposal facilities.

made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

The standard for radiation dose to the general population reflects the standard dose methodology in use at the time 10 CFR 61 was promulgated, which was International Commission on Radiological Protection (ICRP)-2 (ICRP 1959). The TFF PA performance objective is stated as "doses to the representative members of the public shall not exceed 25 mrem (0.25 mSv) in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and progeny in air." The performance measure of 25 mrem in a year total effective dose equivalent (TEDE) reflects the standard dose methodology currently in use, ICRP-30 (ICRP 1979). The measure of protection, 25 mrem, is the same.

The NRC Performance Assessment Working Group (NRC 2000) recommends using ICRP-30 dose methodology in computing potential dose from a LLW disposal facility to compare to this performance objective. The group stated, "that as a matter of policy, the Commission considers 0.25 mSv/yr (25 mrem/yr) TEDE to be an appropriate dose limit to compare with the range of potential doses represented by the older whole body dose limits." Therefore, the performance objective for the general population used in the TFF PA is in agreement with 10 CFR 61.41.

Because 10 CFR 61.41 also has limits for doses to the thyroid and other key organs, these doses are calculated for this draft 3116 Determination. Using conversion factors specified by the EPA in *Limiting Values of Radionuclides Intake and Air Concentrations and Dose Conversion Factors for Inhalation, Submersion, and Ingestion* (EPA 1988), annual doses to the thyroid and other critical organs are calculated for each relevant radionuclide on an organ-by-organ basis. The peak annual dose to the thyroid is approximately 6 mrem/yr compared to the 10 CFR 61.41 limit of 75 mrem/yr. The peak annual dose to any other organ is approximately 0.15 mrem/yr compared to the 10 CFR 61.41 limit of 25 mrem/yr.

The NRC performance objectives of 10 CFR 61.41 also state that reasonable efforts should be made to maintain releases of radioactivity in effluents to the general environment to ALARA. The TFF PA was conducted in accordance with a similar DOE performance objective, which, as quoted in the PA, states "performance assessments shall include a determination that projected releases of radionuclides to the environment shall be maintained as low as reasonably achievable." As described in Section 5 of this document and the TFF PA (DOE-ID 2003b), tank closure (i.e., disposal) activities are being conducted in a manner that will maintain releases of radioactivity ALARA.

The TFF PA presents a detailed description of the modeling approach and computer codes used in the assessment (DOE-ID 2003b) and includes an uncertainty analysis that considers best, realistic, conservative, and worst-case exposure scenarios. The results show that the general population is protected in scenarios used in the PA (DOE-ID 2003b).

Radionuclides released from the INTEC TFF to the environment have the potential to impact humans through a number of different pathways. Radionuclide transport from TFF waste to the aquifer is considered in the groundwater exposure pathway.[ll] Other pathways that may contribute to human exposure are those tied to groundwater concentrations of contaminants. Irrigation with contaminated groundwater may lead to contamination of agricultural crops and animals. Ingestion of contaminated

ll. Three radionuclides contribute to the drinking water pathway dose: ^{129}I, ^{99}Tc, and ^{90}Sr. The maximum dose for groundwater is 0.16 mrem/yr, which occurs at 890 years from ^{129}I. (At 890 years, dose contribution from the remaining radionuclides is negligible.)

groundwater by terrestrial animals may lead to human exposure. In addition, human exposure may occur from ingestion of contaminated groundwater and from consumption of contaminated food supplies (DOE-ID 2003b). Additional pathways, such as the atmospheric transport pathway, were assessed; they provide approximately 0.5 mrem/yr dose to members of the public. As a result, all-pathways dose calculations are based solely on the groundwater pathways.

As Table 19 demonstrates, the PA results for the all-pathways dose to a future member of the public as a function of time after closure indicate a peak dose of 0.46 mrem/yr at approximately 14,000 years after closure. The majority of the total dose is from ^{99}Tc. This projected dose satisfies the 10 CFR 61.41 requirements.

Table 19. Groundwater all-pathways dose (less than 25 mrem/yr).

Nuclide	Decontaminated Tank Residual Inventory (mrem/yr)	Time of Peak Dose (yr)
^{90}Sr/^{90}Y	2.10E−08	551
^{99}Tc	0.46	14,590
^{129}I	0.29	890
Peak Dose	0.46	

To understand parameter sensitivity and to obtain an uncertainty analysis on the TFF PA (DOE-ID 2003b) groundwater model, the INL Site developed worst, conservative, expected, and best-case scenarios to include radionuclide inventory and transport parameters. Since the radionuclide inventory in a 30,000-gal tank is small when compared with a 300,000-gal tank, the impact from the 30,000-gal tanks is small. Data from these four scenarios are used to analyze the probability of radionuclide releases to the public and the environment. This sensitivity/uncertainty analysis shows that radionuclide releases are not under-predicted. The conservative-case scenario is used as the compliance-case scenario in the TFF PA. These evaluations are documented in the TFF PA (DOE-ID 2003b).

7.11 10 CFR 61.42, "Protection of Individuals from Inadvertent Intrusion"

Provisions in 10 CFR 61.42 require:

> [D]esign, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

This requirement reflects the NRC's intent that persons inadvertently intruding into the waste be protected. The performance objective does not place quantitative limits on exposure. However, the environmental impact statement for 10 CFR 61 (NRC 1982) suggests a dose limit of 500 mrem/yr for the

waste classification scheme in 10 CFR 61.55. Consequently, NRC uses a 500-mrem dose limit for evaluating impacts to an inadvertent intruder for purposes of 10 CFR 61.42.[mm]

The PA provides a performance objective for assessing impacts on a person inadvertently intruding into the disposal facility that is more stringent than the NRC requirement. The PA performance objective for inadvertent intruders is:

> [F]or purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the PA shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the [LLW] disposal facility. For intruder analyses, institutional controls shall be assumed to be effective for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute-exposure scenarios, 500 mrem (5 mSv).[nn,oo]

The intruder scenarios and associated doses for the INTEC TFF are evaluated in the PA (DOE-ID 2003b). A site-specific radiological PA is used to verify that requirements for protecting the inadvertent intruder were satisfied. The environmental setting and the logic for using site-specific scenarios is explained in detail in the PA. As described in the PA, several intruder scenarios were evaluated for applicability to the INTEC TFF closure. However, the TFF is a unique environment for intruder scenario evaluations (DOE-ID 2003b). Many of the standard scenarios are not considered applicable to the TFF because the depth of the waste in the tanks is greater than 10 m (33 ft). Therefore, only the intruder-drilling and post-drilling scenarios are considered applicable for the tank and sandpad contamination. Additional scenarios were evaluated for the piping inventory located less than 3 m (10 ft) below the surface. These scenarios include the intruder-construction scenario and the intruder post-construction scenario.

The groundwater pathway was not included in the intruder scenarios because the groundwater pathways dose assessment was performed as part of compliance for protection of the general population from releases of radioactivity. However, if the groundwater all-pathways dose is included with the chronic intruder-drilling scenario it adds 1.35 mrem/yr to the 91.1 mrem/yr if the PA dose assessment is used. The total dose would increase to 92.5 mrem/yr or an increase of 1.5%. If the inventory after cleaning is used, the all-pathways dose is estimated at 0.46 mrem per year and the chronic intruder-drilling scenario adds 25 mrem/yr. The total dose would increase to 25.46 mrem/yr or an increase of 1.8%. The groundwater pathway was prepared to estimate the radiation dose at the location and time that provided the greatest dose to the receptor. The time and location of the greatest drinking water dose (890 years) and location (600 m [1,969 ft]) south of the TFF do not correspond to the time or location for intruder scenarios. The intruder scenario was located directly over the TFF at 100 years after closure, or 2112. The time period for the intruder scenario was chosen because it predicts the greatest dose to intruders from those radionuclides that contribute the majority of the dose. The radionuclide ^{137}Cs and its

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mm. The assessment of impacts to inadvertent intruders need not be a part of the PA required to provide reasonable assurance that the dose limit in 10 CFR 61.41 will not be exceeded. Rather, the site must show that it is following the precepts of the waste classification scheme (i.e., stabilizing Class B waste and providing an intruder barrier for Class C waste) (NRC 1987a, 1987b).

nn. The NRC waste classification system is based on intruder calculations using a 500-mrem dose limit. The TFF PA meets the more stringent DOE guidance for protection of the inadvertent intruder.

oo. The PA uses active institutional controls. Passive controls are not considered, although such controls may be adopted in the future as site closure designs mature. This plan to use active controls without passive controls applies to the TFF only. Both active and passive controls may be used at other sites.

daughter ^{137}Ba contribute over 80% of the dose. ^{137}Cs has a half-life of 30.17 years; therefore, the radiation dose from ^{137}Cs approaches 0 at approximately 250 years. In contrast to the intruder scenario, the groundwater pathway does not increase to the maximum dose until 890 years in the future (Portage 2005m).

For acute exposures, three potential intruder-exposure scenarios are considered: intruder-construction, intruder-discovery, and intruder-drilling scenarios. The intruder-discovery scenario is evaluated to analyze exposures from unearthing equipment and piping nearer than 3 m (10 ft) to the surface. The intruder is assumed to recognize that he or she is digging into very unusual soil immediately upon encountering the vault/tank/piping system and leaves the site. The intruder-discovery scenario is not considered for further analysis because the exposure time is low compared with exposures from the intruder-construction scenario. In the intruder-construction scenario, the intruder is assumed to dig a basement. This scenario is retained for evaluation because of contaminated TFF piping that is less than 3 m (10 ft) below the surface. The maximum dose after institutional control (i.e., 100 years) for the acute intruder-construction scenario is 0.23 mrem. The tank inventory used for the intruder construction scenario is Tank WM-182 because it has the greatest inventory of the cleaned tanks. This total dose is well within the performance objective of 500 mrem for all times after closure. Table 20 shows the major radionuclide contributors and associated dose.

Table 20. Acute intruder-construction dose (less than 500 mrem).

Nuclide	Residual Inventory at Closure (mrem)
^{90}Sr/^{90}Y	0.00005
^{94}Nb	0.001
^{137}Cs/^{137m}Ba	0.20
^{237}Np	0.0001
^{238}Pu	0.009
^{239}Pu	0.006
^{240}Pu	0.003
^{241}Am	0.001
Total Dose	0.23

The credible acute intruder-drilling scenario assumes short-term exposure of a hypothetical intruder to drill-cuttings from a borehole that has penetrated the closed facility and associated waste residuals. The intruder-drilling scenario takes no credit for the thick grout encapsulating the waste but assumes that concentrated waste residuals are spread over a 0.5-acre lot. The exposure pathways analyzed are inhalation of resuspended drill cuttings, external exposure to the ground source, and inadvertent soil ingestion. The acute scenario dose for the limiting tank (represented by the conservative tank residual inventory for Tank WM-185) is approximately 152 mrem at 100 years after closure. Tank WM-185 is used for the chronic drilling scenario because the tank system includes a contaminated sandpad. The drilling scenario assumes the intruder will drill directly through the tank residual and sandpad. This total dose is well within the limit of 500-mrem TEDE. Table 21 lists the major radionuclides contributing to this dose.

Table 21. Acute intruder-drilling dose (less than 500 mrem) (Portage 2005m).

Nuclide	Residual Inventory at Closure (mrem)
$^{90}Sr/^{90}Y$	0.3
^{94}Nb	0.14
$^{137}Cs/^{137m}Ba$	108
^{237}Np	0.07
^{238}Pu	7.71
^{239}Pu	15.1
^{240}Pu	2.82
^{241}Am	17.7
Total Dose	152

For chronic exposures, five intruder scenarios are investigated: intruder post-construction, intruder-resident, intruder-radon, biointrusion, and post-drilling. Again, because of the depth of the final waste form in the tank bottom and piping, intruder post-construction and post-drilling exposure are the only credible scenarios. The chronic intruder post-construction scenario is an extension of the acute intruder-construction scenario. This scenario assumes that an intruder lives in the building constructed as part of the intruder-construction scenario and engages in agricultural activities on the contaminated site. The intruder is exposed to contamination by inhalation of resuspended contaminated soil, inhalation of gaseous radionuclides released from the waste, external irradiation, ingestion of contaminated soil, ingestion of contaminated beef and milk, and ingestion of contaminated vegetables. The maximum dose is 0.23 mrem/yr for the chronic intruder post-construction scenario at 100 years post-closure. This result is well below the performance objective of 500 mrem/yr. Table 22 shows the major radionuclide contributors to the total chronic intruder dose (Portage 2005m).

Table 22. Chronic intruder-construction dose (Portage 2005m).

Nuclide	Residual Inventory at Closure (mrem/yr)
$^{90}Sr/^{90}Y$	0.00005
^{94}Nb	0.001
$^{137}Cs/^{137m}Ba$	0.2
^{237}Np	0.0001
^{238}Pu	0.009
^{239}Pu	0.006
^{240}Pu	0.003
^{241}Am	0.001
Total Dose	0.23

The chronic post-drilling scenario is a logical extension of the drilling scenario considered for acute exposures. This scenario assumes that the intruder occupies the site after drilling a water well and that the intruder grows crops on a mixture of clean soil and contaminated drill cuttings. Exposures from inhaling dust from farming activities, ingesting vegetables grown in the soil, and ingesting products from animals that have eaten contaminated feed are considered. The chronic scenario dose for Tank WM-185 is approximately 25 mrem/yr at 100 years after closure, as shown in Table 23. Tank WM-185 is used for the chronic drilling scenario because the tank system includes a contaminated sandpad. This scenario includes

drilling through the tank residual and sandpad. The short-lived $^{137}Cs/^{137m}Ba$ provides 85% of the total dose. Chronic intruder doses are reduced dramatically with time because of the decay of ^{137}Cs (Portage 2005m).

Table 23. Chronic intruder-drilling dose comparison (Portage 2005m).

Nuclide	Residual Inventory at Closure (mrem/yr)
$^{90}Sr/^{90}Y$	1.69
^{94}Nb	0.03
$^{137}Cs/^{137m}Ba$	21
^{237}Np	0.007
^{238}Pu	0.33
^{239}Pu	0.61
^{240}Pu	0.12
^{241}Am	0.75
Total Dose	25

The uncertainty in the intruder analyses is largely based on the assumptions and parameters selected for the mathematical modeling. However, every attempt was made while running the models to consider the site-specific environment and habits of the people currently in the region. Therefore, the analyses are considered to be representative of the INL Site region. The predicted doses are considered to be conservative, and there is a reasonable assurance that the performance objectives will not be exceeded for the 1,000-year post-closure period.

7.12 10 CFR 61.43, "Protection of Individuals during Operations"

Requirements in 10 CFR 61.43 state:

> [O]perations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter [10 CFR], except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by §61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.

This requirement references 10 CFR 20, "Standards for Protection Against Radiation," which contains radiological protection standards for workers and the public. The DOE requirements for occupational radiological protection are provided in 10 CFR 835, "Occupational Radiation Protection," and those for radiological protection of the public and the environment are provided in DOE Order 5400.5, "Radiation Protection of the Public and the Environment" (1993).

The closure activities at the TFF are maintained under the DOE dose limits for the worker, public, and environment. The TFF PA does not address the operational (i.e., closure and disposal activities) of the facility, only the post-closure (i.e., post-disposal) aspects of the facility. However, during Tank WM-182 cleaning operations, the total radiation exposure to the 23 workers was about 650 mrem for an average exposure of about 30 mrem per individual. This information is based upon a review of dosimetry results from TFF radiation work permits for the January 2002 to June 2005 timeframe (Martin 2005), and the following is summarized as further explained in Appendix B-4:

- The average radiation exposure per tank that will be experienced for cleaning each TFF tank is expected to total about 650 mrem for all occupational exposure

- The average exposure per person for cleaning a TFF tank will be about 650 mrem divided by 23 people, which is about 30 mrem per person

- The maximum radiation exposure for an individual worker is estimated to be 120 mrem for cleaning a single TFF tank.

Worker dose for tank cleaning is minimal because all cleaning is accomplished remotely. Worker exposures would be limited to equipment installation and operation and maintenance activities on contaminated equipment. Worker exposure per tank is estimated to total approximately 650 mrem for 23 workers (30 mrem per worker), which results in a total exposure of about 7.15 rem for cleaning 11 300,000-gal tanks. Therefore, based on estimates that future tank cleaning will probably be accomplished in 1 year and that minimal exposure is expected during grouting (INEEL 1998), an estimated worker dose of 30 mrem/yr is reasonable.

Consistent with Section 3116(a), the cross-referenced "standards for radiation protection" in 10 CFR 20 that are considered in detail in this draft 3116 Determination are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 20.1201(a)(1)(i), 20.1201(a)(1)(ii), 20.1201(a)(2)(i), 20.1201(a)(2)(ii), 20.1201(e), 20.1208(a), 20.1301(a)(1), 20.1301(a)(2), and 20.1301(b).[pp] These dose limits correspond to the dose limits in 10 CFR 835 and relevant DOE orders that establish DOE regulatory and contractual requirements for DOE facilities and activities. The following subsections show that closure operations of the TFF meet these dose limits and that doses will be maintained ALARA.[qq]

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pp. The introductory "notwithstanding" phrase to Section 3116 makes it clear that the provisions of Section 3116(a) are to apply in lieu of other laws that "define classes of radioactive waste." As is evident from the plain language of this introductory "notwithstanding" phrase, Section 3116(a) pertains to classification and disposal, and radiation protection standards for disposal, of certain waste at certain DOE sites. Thus, the factors for consideration set forth in Subsections (a)(1)–(3) of Section 3116 are those that pertain to classification and disposal of waste, and the radiation protection standards for disposal. The Joint Explanatory Statement of the Committee of Conference in Conference Report 108-767, accompanying H.R. 4200 (NDAA), also confirms that 3116(a) concerns classification, disposal, and radiation protection standards associated with disposal, and does not concern general environmental laws or laws regulating radioactive waste for purposes other than disposal. Moreover, in the plain language of Section 3116, Congress directed that the Secretary of Energy consult with the NRC but did not mandate that DOE obtain a license or any other authorization from NRC, and did not grant NRC any general regulatory, administrative, or enforcement authority for disposal of the DOE wastes covered by Section 3116. As such, the "standards for radiation protection" in 10 CFR 20 (as cross-referenced in the performance objective in 10 CFR 61.43), which are relevant in the context of Section 3116, are the dose limits for radiation protection of the public and the workers during disposal operations, and not those which address general licensing, administrative, programmatic, or enforcement matters administered by NRC for NRC licensees. Accordingly, this draft 3116 Determination addresses in detail the radiation dose limits for the public and the workers during disposal operations that are contained in the provisions of 10 CFR 20 referenced above. Although 10 CFR 20.1206(e) contains limits for planned special exposures for adult workers, there will not be any such planned special exposures for closure operations at TFF. Therefore, this limit is not discussed further in this draft 3116 Determination. Likewise, 10 CFR 20.1207 specifies occupational dose limits for minors. However, there will not be minors working at TFF who will receive an occupational dose. Therefore, this limit is not discussed further in this draft 3116 Determination.

qq. In addition, 10 CFR 835, like 10 CFR 20 for NRC licensees, includes requirements that do not set dose limits, such as requirements for radiation protection programs, monitoring, entrance controls for radiation areas, posting, records, reporting, and training.

7.12.1 Air Emissions Limit for Individual Member of the Public [10 CFR 20.1101(d)]

The NRC regulation in 10 CFR 20.1101(d) provides in relevant part:

> [A] constraint on air emissions of radioactive material to the environment, excluding radon-222 and its daughters, shall be established … such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv)/yr from these emissions.

The DOE similarly limits doses from air emissions to the public to 10 mrem/yr in DOE Order 5400.5 (1993). The DOE is also subject to and complies with the EPA's requirement in 40 CFR 61.92, which has the same limit.[rr] The estimated dose per year from airborne emissions to the maximally exposed individual member of the public resulting from closure/disposal activities is 0.51 mrem/yr (DOE-ID 2003b).

7.12.2 Total Effective Dose Equivalent Limit for Adult Workers [10 CFR 20.1201(a)(1)(i)]

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> (1) An annual limit, which is the more limiting of –
>
> (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv).

The DOE's regulation in 10 CFR 835.202(a)(1) has the same annual dose limit for the annual occupational dose to general employees.[ss] For the occupational dose to adults during closure operations at the TFF, the TEDE per year will be controlled ALARA below 5 rem, as shown in Appendix B-4.

7.12.3 Any Individual Organ or Tissue Dose Limit for Adult Workers [10 CFR 20.1201(a)(1)(ii)]

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

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rr. 40 CFR 61.92 provides as follows: "Emissions of radionuclides to the ambient air from DOE facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr. It is assumed that the individual is an adult living at the site perimeter that is exposed to the maximum yearly radioactive atmospheric release and maximum radiation concentration in food for 365 days per year. For the airborne pathway, the dose is developed by the input of atmospheric release data, vegetation consumption data, milk consumption data, and beef consumption data."

ss. The DOE's regulation requires that the occupational dose per year for general employees shall not exceed both a TEDE of 5 rem and the sum of the deep-dose equivalent for external exposures and the committed dose equivalent to any other organ or tissue other than the lens of the eye of 50 rem. The NRC's regulation specifies that either of these two limits shall be met by NRC licensees, whichever is more limiting. This draft 3116 Determination shows that DOE will meet the more stringent of the dose limits in 10 CFR 835 and the relevant dose limits in 10 CFR 20. Because DOE imposes stricter, separate requirements, the provisions of 10 CFR 20.1201(a)(1) and (a)(2), which correlate to 10 CFR 835.202(a)(1) and (a)(2), are discussed in separate subsections in this draft 3116 Determination.

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> (1) An annual limit, which is the more limiting of –…
>
> (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

The dose limit specified in 10 CFR 20.1201(a)(1)(ii) is the same as that specified in 10 CFR 835.202(a)(2). For the occupational dose to adults during disposal/closure operations at TFF, the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye will be controlled to ALARA below a maximum of 50 rem/yr. The INL "Radiological Control Manual" (PRD-183, 2002) provides that the design basis annual occupational exposure limits for any organ or tissue other than the eye cannot exceed 50 rem/yr, which is the same as the NRC limit. Furthermore, TFF disposal/closure operations will consist predominantly of remote cleaning technologies to minimize hands-on work and associated exposure.

7.12.4 Annual Dose Limit to the Lens of the Eye for Adult Workers [10 CFR 20.1201(a)(2)(i)]

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposure the following dose limits. …
>
> (2) The annual limits to the lens of the eye, to the skin of the whole body or to the skin of the extremities, which are:
>
> (i) A lens dose equivalent of 15 rems (0.15 Sv).

The dose limit specified in 10 CFR 20.1201(a)(2)(i) is the same as that specified in DOE's regulation in 10 CFR 835.202(a)(3). For the occupational dose to adults during closure operations at the TFF, the annual dose limit to the lens of the eye will be controlled to ALARA below a maximum of 15 rem/yr. Furthermore, disposal/closure operations will consist predominantly of remote cleaning technologies to minimize hands-on work and associated exposure.

7.12.5 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers [10 CFR 20.1201(a)(2)(ii)]

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits. …
>
> (2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are: …

> (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

This NRC dose limit specified in 10 CFR 20.1201(a)(2)(ii) is the same as the DOE dose limit specified in 10 CFR 835.202(a)(4). For the occupational dose to adults during closure operations at the TFF, which involve little hands-on activity, the annual dose limit to the skin of the whole body or to the skin of any extremity will be controlled to ALARA below a shallow-dose equivalent of 50 rem/yr. Furthermore, disposal/closure operations will consist predominantly of remote cleaning technologies to minimize hands-on work and associated exposure.

7.12.6 Limit on Soluble Uranium Intake [10 CFR 20.1201(e)]

The NRC regulation in 10 CFR 20.1201(e), concerning occupational dose limits for adults, provides in relevant part: "in addition to the annual dose limits, … limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity."

Requirements in DOE Order 440.1A (1998) for soluble uranium intake are the more restrictive of the concentrations in the American Conference of Governmental Industrial Hygienists Threshold Limit Values (0.2 mg/m^3, which is the same as noted in 10 CFR 20, Appendix B) or the Occupational Safety and Health Administration permissible exposure limit (0.05 mg/m^3) (DOE O 440.1A, 1998). The permissible exposure limit for soluble uranium, which equates to a soluble uranium intake of 2.4 mg/week, is the more restrictive of the two. Therefore, this limit is imposed for TFF closure operations. Furthermore, disposal/closure operations will consist predominantly of remote cleaning technologies to minimize hands-on work and associated exposure.

7.12.7 Dose Equivalent to an Embryo/Fetus [10 CFR 20.1208(a)]

The NRC regulation in 10 CFR 20.1208(a), concerning the dose equivalent to an embryo/fetus, provides in relevant part: "ensure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv)."

The DOE's regulation in 10 CFR 835.206(a) has the same dose limit. For the occupational dose to an embryo/fetus during closure operations at the TFF, doses will be controlled so that the dose equivalent to the embryo/fetus during the entire pregnancy for a declared pregnant worker will not exceed 0.5 rem. Furthermore, after declaration of pregnancy, DOE provides the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. In addition, personnel dosimetry is provided and used to track exposure carefully.

7.12.8 Dose Limits for Individual Members of the Public [10 CFR 20.1301(a)(1)]

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

> (a) [C]onduct operations so that –
>
> (1) The total effective dose equivalent to individual members of the public …does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to

> individuals administered radioactive material and released…, from voluntary participation in medical research programs, and from the …disposal of radioactive material into sanitary sewerage.

Provisions in DOE Order 5400.5(II.1.a) (1993) similarly limit public doses to less than 100 mrem/yr. However, DOE's application of the limit is more restrictive in that DOE is required to make a reasonable effort to ensure that multiple sources (e.g., DOE sources and NRC-regulated sources) do not combine to cause the limit to be exceeded. For individual members of the public during closure operations at the TFF, the TEDE limit to an individual member of the public will be controlled to less than 0.1 rem/yr. The dose to the maximally exposed individual outside the INL Site boundary is calculated each year and shown less than 100 mrem/yr. The latest annual report shows the dose to be 0.035 mrem/yr for Calendar Year 2003 (DOE-ID 2004a).

7.12.9 Dose Limits for Individual Members of the Public [10 CFR 20.1301(a)(2)]

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

> (a) [C]onduct operations so that – …
>
> > (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released …, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

The DOE's regulation in 10 CFR 835.602 establishes the expectation that the TEDE in controlled areas will be less than 0.1 rem/yr. To ensure that these dose limits are met, the following measures have been instituted within controlled areas. Per 10 CFR 835.603(g), radioactive materials areas have been established for accumulations of radioactive material that could result in a radiation dose of 100 mrem/yr or greater. Averaged over a work year, this yields a constant dose rate of 0.00005 rem/hr as compared to the NRC limit of 0.002 rem in any one hour. In addition, training and dosimetry are required for individual members of the public for entry into controlled areas.

7.12.10 Dose Limits for Individual Members of the Public [10 CFR 20.1301(b)]

The NRC regulation in 10 CFR 20.1301(b), concerning dose limits for individual members of the public, provides in relevant part: "if … members of the public [are permitted] to have access to controlled areas, the limits for members of the public continue to apply to those individuals."

The DOE's regulation in 10 CFR 835.208 has the same dose limit. The TEDE limit to an individual member of the public granted access to controlled areas[tt] during disposal operations at TFF will be controlled to 0.1 rem/yr. Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits,

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tt. 10 CFR 20.1003 defines restricted areas as "an area, access to which is limited … for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials." This definition is the same as that in 10 CFR 835.2 for a controlled area.

use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem/yr.[uu]

7.12.11 As Low As Reasonably Achievable (10 CFR 20.1003)

The NRC regulation in 10 CFR 20.1003 defines ALARA in relevant part: "ALARA … means making every reasonable effort to maintain exposures to radiation as far below the dose limits … as is practical consistent with the purpose for which the … activity is undertaken."

The DOE has a similar requirement, and DOE's regulation in 10 CFR 835.2 defines ALARA as "the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable." For radiological work activities during closure operations at the TFF, every reasonable effort will be made to maintain exposures to radiation as far below the dose limits as is practical consistent with the purpose for which the activity is undertaken. Furthermore, DOE's regulation in 10 CFR 835.101(c) requires the contents of each radiation protection program (RPP) to include formal plans and measures for applying the ALARA process to occupational exposure.

The ALARA process provides for rigorous reviews. Worker dose reduction is achieved through reviews of potential worker exposures against established criteria and then applying appropriate radiological engineering measures, including:

- Inclusion of radiological control points
- Use of work processes and special tools to reduce exposure time
- Use of engineered controls, including fixatives or covering work areas
- Specification of special radiological training or unique monitoring requirements
- Use of mockup training for first of a kind, high exposure, or complex tasks
- Engineering design and use of temporary shielding to reduce radiation levels and time in radiation fields
- Walk-down or dry run of activity using applicable procedures
- Staging and prefabrication of necessary materials and special tools
- Maximization of prefabrication and shop work
- Provisions for waste minimization.

The institutionalized application of the ALARA program for TFF activities contributed to the low doses achieved for previous tank work as described in Appendix B-4.

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uu. 10 CFR 20.1301(d) allows licensees to request NRC authorization to allow an individual member of the public to receive up to an annual dose limit of 0.5 rem. 10 CFR 835 is more restrictive for the dose to an individual member of the public with a limit of 0.1 rem maximum annual dose.

7.12.12 Reasonable Assurance

Measures that provide reasonable assurance that closure operations at the TFF will comply with the applicable dose limits and with the ALARA provisions include: (1) the documented RPP; (2) the safety analysis; (3) design; (4) regulatory and contractual enforcement mechanisms; and (5) access controls, training, and dosimetry. These measures are discussed in the following subsections.

7.12.12.1 INL Radiation Protection Program. The DOE regulates occupational radiation exposure at its facilities through 10 CFR 835, "Occupational Radiation Protection," which establishes exposure limits and other requirements to ensure that DOE facilities are operated in a manner such that occupational exposure to workers is maintained within acceptable limits and as far below these limits as is reasonably achievable. The requirements in 10 CFR 835 are nuclear safety requirements, which if violated, provide a basis for the assessment of civil penalties under the Section 234A of the Atomic Energy Act (42 USC 2011 et seq., 1954).

Pursuant to 10 CFR 835, activities at the INL Site, including closure operations at the TFF, must be conducted in compliance with the documented RPP for the INL Site as approved by DOE (PLN-260, 2003). The key elements of the RPP include monitoring of individuals and work areas, control of access to areas containing radiation and radioactive materials, use of warning signs and labels, methods to control the spread of radioactive contamination, radiation safety training, objectives for the design of facilities, criteria for levels of radiation and radioactive material in the workplace, and continually updated records to document compliance with the provisions of 10 CFR 835. The RPP also includes formal plans and measures for applying the ALARA process.

The RPP procedures control the planning of radiological work, the use of radiation monitoring devices by employees, the bioassay program, the air monitoring program, the contamination control program, the ALARA program, the training of general employees, radiological workers, radiological control inspectors, and health physics professionals and technicians, and the other aspects of an occupational RPP as required by 10 CFR 835.

7.12.12.2 Documented Safety Analysis. The DOE has approved a safety analysis report for operating the TFF in accordance with 10 CFR 830. The "Safety Analysis Report for the Tank Farm Facilities" (SAR-107, 2004) identifies, classifies, and evaluates hazards associated with the closure project. The first step in developing the safety analysis report was a formal hazards analysis to evaluate the potential risk of operations to the workers and the public. A group of subject-matter experts with expertise in operations, engineering, industrial hygiene, radiological protection, environmental compliance, and maintenance performed the analysis.

The hazards analysis consisted of three basis phases: hazard identification, hazard classification, and hazard evaluation. During the hazard identification phase, all possible radiological and chemical hazardous materials associated with the normal and abnormal operations of the facility were identified, along with all potential energy sources available to disperse the hazardous materials to the environment.

During the hazard classification phase, the maximum quantities of hazardous materials possible in the TFF were evaluated against the criteria listed in DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports" (1997), to determine the overall hazard classification of the facility. The team determined that the hazard classification of the TFF was Hazard Category 2. This classification denotes a potential for significant onsite consequences. This facility has the potential for release of sufficient quantities of hazardous material, which would require onsite emergency planning activities.

During the third and final phase of the hazard analysis, all possible normal and abnormal operational events that could result in exposing facility workers or the public to hazardous material were evaluated to determine the magnitude of the risk. During this hazard evaluation phase, the consequence and frequency of each operational event were determined qualitatively, and the resulting level of risk was identified. The purpose of identifying the level of risk was to determine which operational events posed some level of risk (and thus required additional evaluation) and which events presented negligible risk to the facility workers and public.

The safety analysis report analyzed the hazards that were identified in the hazards analysis that could impact facility workers during normal operations and accident conditions, and specifically included radiation exposure hazards. The report identified the basis for derivation of the TFF technical safety requirements and also provided summary descriptions of the key features of safety management programs at the INL Site as they pertain to the TFF (see Appendix D).

7.12.12.3 Regulatory and Contractual Enforcement. Any violation of the requirements in 10 CFR 835 is subject to civil penalties pursuant to Section 234A of the Atomic Energy Act of 1954 (42 USC 2011 et seq., 1954), as implemented by DOE regulations in 10 CFR 820. In addition, the requirements in 10 CFR 835 and all applicable DOE orders are incorporated into all contracts with DOE contractors, the DOE contractor for closure operations at the TFF, as well as other operations at the INL Site. The DOE enforces these contractual requirements through contract enforcement measures, including the reduction of contract fees.

7.12.12.4 Access Controls, Training, Dosimetry, and Monitoring. Training or an escort is required for individual members of the public to enter controlled areas. In addition, dosimetry is required if a member of the public is expected to enter a controlled area and exceed a dose of 0.05 rem/yr to ensure radiation exposure limits are not exceeded for any member of the public.

Also, worker radiation exposure monitoring is performed for all workers expected to receive 100 mrem/yr from internal and external sources of radiation to provide that no worker exceeds radiation exposure limits and that all radiation doses are maintained as far below the limits as is reasonably achievable.

7.13 10 CFR 61.44, "Long-Term Stability of Disposal Site"

10 CFR 61.44 states:

> The disposal facility must be sited, designed, used, operated and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

As discussed previously, several of the tanks in the TFF have been removed from service and cleaned in preparation of final closure. Once the remaining SBW has been transferred from the tanks for treatment and disposal outside the State of Idaho, the remainder of the tanks will be cleaned and closed. Cleaning and closure of the TFF is planned to be completed by 2012. As shown below, disposal of the solidified TFF residual waste and TFF tank system in situ will meet the performance objectives of 10 CFR 61.44 for long-term stability.

7.13.1 Siting

A comprehensive review of site geology, seismology, hydrology, demography, meteorology, and environmental setting are presented in Section 2 of the TFF PA (DOE-ID 2003b), which is briefly summarized as follows. The INL is located in southeastern Idaho, on the west-central portion of the Eastern Snake River Plain (ESRP). The nearest INL boundaries are 35 km (22 mi) west of Idaho Falls, 37 km (23 mi) northwest of Blackfoot, 71 km (44 mi) northwest of Pocatello, and 11 km (7 mi) east of Arco. There are no permanent residents within an 18-km (11-mi) radius of INTEC. The Snake River Plain Aquifer is approximately 122 m (400 ft) below ground surface at INTEC.

The principal surface materials at the INL are basalt, alluvium, lakebed sediments, volcanic rocks, and sedimentary rocks. The natural plant life consists mainly of sagebrush and various grasses. The most prominent ground cover is a mixture of sagebrush and a variety of grasses. The soil at the INL TFF site is previously disturbed sandy gravel; the flat terrain precludes erosion.

The seismically active intermountain seismic belt surrounds the ESRP. The intermountain seismic belt is a zone of concentrated seismicity, which extends from northwestern Montana to eastern Idaho. The ESRP is not a part of the intermountain seismic belt, and therefore, is not seismically active. The primary seismic hazards from earthquakes to INL Site facilities consist of the effects from ground shaking and surface deformation. Based on the seismic history and geologic conditions, earthquakes greater than a magnitude of 5.5 and associated strong ground shaking and surface fault ruptures are not likely to occur within the ESRP (DOE-ID 2003b).

A large earthquake, in the vicinity of the INL but outside the ESRP, occurred in 1983 and had a surface-wave magnitude of 7.3. The epicenter for this event was located approximately 89–97 km (55–60 mi) from the INL Site. Although the earthquake ground motions were felt at the INL Site, only minor non-nuclear building damage occurred in the form of hairline cracks and settlement. The INTEC facility did not experience structural failures or waste spills as a result of the earthquake and waste storage facilities did not show evidence of permanent movement or resulting damage. Peak ground accelerations ranging from 0.022 to 0.078 g were recorded at several INL Site facility areas. The largest earthquake in the region occurred in, 1959, at Hebgen Lake, Montana, located approximately 193 km (120 mi) northeast of the INL. The event had a surface-wave magnitude of 7.5; it was felt at the INL Site, but caused no damage (DOE-ID 2003b).

Volcanic hazards include the effects of lava flows, fissures, uplift, subsidence, volcanic earthquakes, and ash flows or airborne ash deposits. Most of the volcanic activity occurred from 4 million to 2,100 years ago in the INL Site area. The most recent and closest volcanic eruption occurred at the Craters of the Moon National Monument, 46 km (27 mi) southwest of INTEC. Volcanic activity is a result of large volumes of magma or hotspots relatively near the Earth's surface. Because the hotspot is now situated beneath Yellowstone National Park and continues to move away from the INL Site, recurrence of volcanic activity in the INL Site area is unlikely (DOE-ID 2003b).

The Big Lost River enters the INL Site near the southern end from the west. During exceptionally wet years, the Big Lost River flows in a large arc north to the foot of the Lemhi Mountain Range, where it ends in a series of playas (sinks). The northwest boundary of INTEC is closest to the Big Lost River channel, approximately 61 m (200 ft) away. The Big Lost River is the principle natural surface water feature on the INL and the only stream with potential impacts to the TFF.

A record snow pack occurred in the Big Lost River Basin in the winter of 1964/1965. This flood is significant because it exhibited the largest crest and largest water volume to be discharged onto the INL Site in 65 years of record, yet caused no damage to INL Site facilities. In a review of the historical

information, no flooding or inundation from storms or runoff has caused flooding of the INTEC site. To evaluate the potentially higher infiltration rates at the TFF, modeling was conducted to simulate a 500-year flood event of the Big Lost River. The analysis performed as part of the PA sensitivity analysis indicated there was no significant increase to the groundwater receptor based on these high transient infiltration rates.

7.13.2 Design

As described in Section 2 of this document, the TFF design provides for 11 underground 300,000-gal stainless tanks surrounded by concrete vault structures. These tanks have a right-cylinder shape, and are approximately 15.2 m (50 ft) in diameter and approximately 6.4–7.0 m (21–23 ft) in height. The bottom and lower half of the tank walls are constructed of 0.8-cm (0.3-in.) thick Type 304L stainless steel with two tanks constructed of Type 307 stainless steel. The upper half to the tank wall and domes are constructed of 0.63-cm (0.25-in.) thick stainless steel. Stainless steel piping of approximately 5 cm (2 in.) in diameter is in place between the tanks to allow transfers to and from individual tanks and process facilities.

The concrete vaults are of three different designs. The vaults surrounding Tanks WM-180 and WM-181 are approximately 0.3 m (1 ft) thick, including a 0.6-m (2-ft) thick vault floor. An approximately 15-cm (6-in.) thick layer of sand was placed between the bottom of the tanks and the vault floor to provide cushioning for the tank. The vaults that surround Tanks WM-182 through WM-186 are of different construction. The concrete walls are approximately 0.3 m (1 ft) in thickness and the floor is approximately 0.6 m (2 ft) in thickness. The vaults surrounding Tanks WM-187 through WM-190 have walls that are 1 m (3 ft) thick and a floor that is 0.8 m (2.5 ft) thick. A minimum space of approximately 0.5–1.2 m (1.5–4 ft) between the outer tank walls and the inside of the vault wall is referred to as the tank annulus. The space and volume of the annulus surrounding the tanks varies because of varied vault construction.

The four 30,000-gal tanks are constructed much differently. These tanks are horizontal cylinders (“cigar-shaped”) approximately 3.5 m (11.5 ft) in diameter and 11.6 m (38 ft) in length. The tanks were made of Type 316L stainless steel (thicker plate than the 300,000-gal tanks (1.7 cm [0.68-in.] cylinder and 0.27-cm [0.5-in.] heads), and had extra plate installed over the weld seams to protect them from corrosion. These tanks are not surrounded by a vault structure. Each of the 30,000-gal tanks is buried approximately 3 m (10 ft) underground and rests on a curbed concrete pad. Stainless steel piping of approximately 5 cm (2 in.) diameter is in place between the tanks allow transfers from individual tanks and to process facilities.

7.13.3 Use/Operation

The TFF is used to store a variety of wastes as described in Section 2 of this document. These wastes include both radioactive constituents and constituents that are considered hazardous under RCRA (42 USC 6901 et seq., 1976). As such, the TFF is operated regulated by the State of Idaho as an interim status unit under HWMA (State of Idaho 1983) regulations. As described in Section 2, seven of the 300,000-gal tanks and the four 30,000-gal tanks have been removed from operational service and cleaned in preparation of future closure. Four other 300,000-gal tanks remain in an active, operational status to store the remaining SBW. In the future, as the SBW is removed from the tanks for treatment and disposal, these tanks will be cleaned, grouted, and closed.

7.13.4 Closure

As described in Section 2, closure of the tanks includes cleaning of the tanks, vaults, and associated interconnecting piping and ancillary equipment, followed by completely filling these components with a grout material to immobilize any remaining residual waste and provide for a stable structure. Current plans for closure of the TFF tanks do not include additional engineered barriers or controls such as a cover system or cap over the TFF.

Long-term stability of the disposal site after closure means that the waste maintains structural integrity under the expected disposal conditions. As such, the long-term stability of the closed facility is an important element of meeting the performance objectives. Stability prevents trench subsidence, water infiltration, and radionuclide release because of disintegration of the waste form, and minimizes the likelihood of intrusion into the waste. The waste form itself can provide structural stability by processing the waste to a stable form or by placing the waste in a disposal container or structure that provides stability after disposal. The primary barrier relied upon at the TFF to provide structural stability for the tank waste residuals is the stainless steel tank. The primary barrier for the residuals in the vault and sandpads is the concrete vault walls and the grouted vault, which provide structural stability and reduce migration of contamination. The primary barrier for the residuals in the transfer piping is the stainless steel piping (DOE-ID 2003b).

The grouted tanks, vaults, piping, and valve boxes provide a long-term, stable waste form. For guidance, DOE considered the *Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility* (NRC 1987a), which notes that site stability is focused on reducing the contact of water with the waste and providing assurance that active maintenance will not be needed following closure. The long-term stability of the facility is evaluated in the TFF PA (DOE-ID 2003b), which provides a degradation analysis of the grouted tanks, vaults, and piping, and shows the grout will likely remain intact for at least 2,000 years, but much more likely to remain intact for much longer. As shown above, the site conditions do not present hazards that impact the stability of the closed TFF. In addition, the methods used to close the TFF will result in a closed facility that does not require ongoing active maintenance of the facility following closure. As such, the performance of the closed TFF complies with the objectives of 10 CFR 61.44.

8. STATE-APPROVED CLOSURE PLANS

Key Points

- The TFF is being closed to meet clean closure standards for hazardous constituents. Clean closure is the removal or decontamination of all waste residues to meet State-approved performance objectives for hazardous constituents. If these clean closure standards are not met, the TFF must be closed as a landfill in accordance with hazardous waste regulations. However, waste removal and decontamination activities needed to meet performance objectives for radioactive constituents will likely result in the removal of hazardous constituents to the clean closure standard.
- Prior to closure, the closure plan for the four 300,000-gal tanks remaining to be cleaned will be submitted to the State of Idaho.
- A final closure plan for all of the TFF tanks will be submitted to the State of Idaho after each partial closure phase of the TFF is completed.

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
>
> > (3)(A)(ii) [will be disposed of] pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section.

As discussed in Subsection 2.3, the TFF closure is being performed under both DOE and State of Idaho Department of Environmental Quality (DEQ) requirements, which includes preparing closure plans. The wastes stored in the TFF are mixed wastes.[vv] As such, the State of Idaho regulates the hazardous constituents and the DOE regulates the radioactive constituents. Because the TFF stores mixed wastes, TFF closure must comply with closure requirements for hazardous waste as well as radioactive waste. For hazardous waste, closure must comply with HWMA (State of Idaho 1983) and RCRA (42 USC 6901 et seq., 1976) as implemented by IDAPA 58.01.05.009 and 40 CFR 265, "Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities." These regulations require the preparation of a closure plan that includes specific elements. Because the TFF tanks do not meet secondary containment standards, the regulations also specify that both a closure plan for clean closure and a contingent closure plan for closure of the TFF as a landfill must be prepared. Clean closure is the removal or decontamination of waste residues to meet State-approved performance objectives for the hazardous constituents. Landfill closure is necessary if the clean closure standards cannot be met. The clean closure standards are included as part of the HWMA/RCRA closure plans. For all of the tanks and ancillary equipment cleaned from 2002 to 2005, these clean closure requirements have been met.

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vv. A mixture of hazardous wastes regulated under Subtitle C of the Resource Conservation and Recovery Act of 1976 (42 USC 6901 et seq., 1976) and radioactive wastes regulated under the Atomic Energy Act of 1954 (42 USC 2011 et seq., 1954).

In accordance with these requirements, the DOE has prepared a series of closure plans to support the TFF closure. These closure plans present the strategy for clean closure of the TFF tanks to site-specific action levels. The closure plan for TFF Tanks WM-187, WM-188, WM-189, and WM-190 will be submitted prior to closure. In addition, a final closure plan for all of the TFF tanks will be prepared and submitted to the state for approval. Final closure actions, such as grout placement, will not be initiated until the DEQ approves the final closure plan.

The following closure plans have been submitted to DEQ for approval:

- Tanks WM-182 and WM-183—*Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks WM-182 and WM-183* (DOE-ID 2003a)

- Tanks WM-184, WM-185, and WM-186—*Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks WM-184, WM-185, and WM-186* (DOE-ID 2004b)

- Tanks WM-103, WM-104, WM-105, WM-106, and WM-181—*Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks WM-103, WM-104, WM-105, WM-106, and WM-181* (DOE Idaho 2004a)

- Tank WM-180—*Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act Closure Plan for Idaho Nuclear Technology and Engineering Center Tank WM-180* (DOE Idaho 2004b).

In addition, the DOE has prepared and submitted to DEQ two contingent landfill closure plans as required by the hazardous waste management regulations. The *Contingent Landfill Closure and Post-Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks WM-182 and WM-183* (DOE-ID 2003e) applies to TFF Tanks WM-182 and WM-183. The *Contingent Landfill Closure and Post-Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks in the Tank Farm Facility* (DOE-ID 2003f) applies to all of the tanks in the TFF. Previously, contingent landfill closure plans were planned to be developed for each closure phase, in a manner similar to the clean closure plans. However, DEQ and DOE agreed that one contingent landfill closure plan should be developed for the entire TFF.

The closure activities presented in these plans are designed to satisfy both DOE radioactive waste requirements and HWMA/RCRA hazardous waste requirements. Although contingent landfill closure plans were prepared as required, the TFF is being closed to HWMA/RCRA clean closure standards. Waste removal and decontamination activities needed to remove the required radionuclides to meet performance objectives for the radioactive waste will result in the removal of the hazardous constituents to the HWMA/RCRA clean closure standard.

In addition to the closure plans, a SAP for each closure phase is submitted to DEQ for approval. These plans describe the sampling and analysis to be performed to confirm that the residual wastes remaining the tanks and ancillary equipment: (1) are not hazardous waste, (2) meet the clean closure action levels specified in the associated HWMA/RCRA closure plan, and (3) meet the DOE performance objectives. As the sampling and analysis activities for a closure phase are completed, a series of DQA reports are prepared. The purpose of the DQAs is to evaluate the quality of the characterization data collected to determine whether the data could be used to meet the DQOs established for the sampling and analysis activities. The following SAPs have been submitted to DEQ:

- Tanks WM-182 and WM-183

  - *Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-182 and WM-183 Tank Residuals* (INEEL 2002b)

  - *Sampling and Analysis Plan for the Post-Decontamination Characterization of the Process Waste Lines from INTEC Tank Farm Facility Tanks WM-182 and WM-183* (INEEL 2001)

- Tanks WM-184, WM-185, and WM-186

  - *Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-184, WM-185, and WM-186 Tank Residuals* (INEEL 2003)

- Tanks WM-103, WM-104, WM-105, WM-106, and WM-181

  - *Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-103, WM-104, WM-105, WM-106, and WM-181 Tank Residuals* (ICP 2004a)

- Tank WM-180

  - *Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-180 Tank Residuals* (ICP 2004b).

The following DQA reports have been provided to DEQ:

- Tanks WM-182 and WM-183

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-182 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (INEEL 2004a)

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-183 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (INEEL 2004b)

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Ancillary Equipment Associated with Tanks WM-182 and WM-183 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004c)

- Tanks WM-184, WM-185, and WM-186

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-184 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004d)

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-185 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004e)

- *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-186 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004f)

- *Data Quality Assessment Report for the Post-Decontamination Characterization of the Ancillary Equipment Associated with Tanks WM-184, WM-185, and WM-186 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004g)

- Tanks WM-103, WM-104, WM-105, WM-106, and WM-181

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tanks WM-103, WM-104, WM-105, and WM-106 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004h)

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-181 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004i)

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Ancillary Equipment Associated with Tanks WM-103, WM-104, WM-105, WM-106, and WM-181 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2004j).

- Tank 180

  - *Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM-180 and Associated Ancillary Equipment at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (ICP 2005b).

As of May 2005, DEQ has approved two of the closure plans for clean closure (DOE-ID 2003a, 2004) and their associated SAPs (INEEL 2001, 2002b, 2003). Both of the contingent landfill closure plans (DOE-ID 2003e, 2003f) have also been approved. Approval of the remaining closure plans and SAPs is pending.

Results of the confirmatory sampling and analysis of the TFF tank waste residuals for the TFF closure activities performed as of May 2005 indicate that HWMA/RCRA clean closure requirements and DOE performance objectives as set forth in the closure plans have been met. The final closure of the TFF will be certified by an independent, Idaho-registered professional engineer, the facility contractor, and/or the DOE Idaho Operations Office, in accordance with IDAPA 58.01.05.009 (40 CFR 265.115).[ww]

ww. Certification of partial closures is not required (EPA 1998).

9. CONCLUSIONS

As shown in the preceding sections of this draft 3116 Determination, the stabilized TFF residuals and TFF tank system are not HLW based on the considerations set forth in Section 3116(a) of the NDAA and may be disposed of as LLW at the INL Site in accordance with Section 3116 of the NDAA. This draft 3116 Determination will be finalized after the DOE has completed consultation with the NRC, and although not required by Section 3116, after public review and comment.

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11. DRAWING

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Appendix A

INTEC Tank Farm Facility Closure
Supporting Tables and Photographs

Appendix A

INTEC Tank Farm Facility Closure
Supporting Tables and Photographs

In support of the Tank Farm Facility (TFF) closure, inventory tables were generated for all tanks that have been cleaned up to this point; these tables are presented is Section A-1. Inventories at closure for all of the tanks that have been cleaned show that Tank WM-182 contains the highest Ci content. The inventory at closure for Tank WM-182 has been compared to the performance assessment (PA) single tank inventory in Section A-2 to show greater than expected removal of radioactivity in the tanks. Post-decontamination photos of the insides of the tanks are provided in Section A-3 to visually show the extent of the decontamination process. Fractions of the Class C concentration limits were calculated for the tanks, including the mass of the steel of the tanks and for the piping after grouting. Section A-4 presents alternative fraction of Class C concentration limits without including the mass of steel for the tanks and the mass of the grout for the piping.

A-1. POST-DECONTAMINATION RADIONUCLIDE INVENTORIES

Table A-1. Post-decontamination estimated inventory for Tank WM-103.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ac	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	4.02E–08	^{227}Ac	1.20E–09	6.28E–06	7.51E–15	3.22E–12	4.02E–08	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ac	1.73E–14	6.28E–06	1.08E–19	4.65E–17	5.81E–13	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	9.59E–12	^{108}Ag	2.85E–13	6.28E–06	1.79E–18	7.68E–16	9.59E–12	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.08E–10	^{108m}Ag	3.20E–12	6.28E–06	2.01E–17	8.63E–15	1.08E–10	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	8.00E–16	^{109m}Ag	2.38E–17	6.28E–06	1.49E–22	6.41E–20	8.01E–16	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	3.01E–17	^{110}Ag	8.93E–19	6.28E–06	5.61E–24	2.41E–21	3.01E–17	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	2.26E–15	^{110m}Ag	6.71E–17	6.28E–06	4.22E–22	1.81E–19	2.26E–15	0.00%
^{241}Am	[d]		3.40E–04	6.36E–03	^{241}Am	[e]		2.55E–11	1.09E–08	6.36E–03	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	2.40E–05	^{242}Am	7.13E–07	6.28E–06	4.48E–12	1.92E–09	2.40E–05	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	2.41E–05	^{242m}Am	7.17E–07	6.28E–06	4.50E–12	1.93E–09	2.41E–05	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	3.31E–05	^{243}Am	9.83E–07	6.28E–06	6.17E–12	2.65E–09	3.31E–05	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	1.89E–10	^{217}At	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{137m}Ba	[d]		9.20E–01	1.72E+01	^{137m}Ba	[e]		6.28E–06	2.69E–03	1.72E+01	47.32%
^{10}Be	7.56E–11	1.8	1.36E–10	2.55E–09	^{10}Be	7.56E–11	6.28E–06	4.75E–16	2.04E–13	2.55E–09	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Bi	1.11E–10	6.28E–06	7.00E–16	3.00E–13	3.75E–09	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	1.82E–22	^{210m}Bi	5.41E–24	6.28E–06	3.40E–29	1.46E–26	1.82E–22	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Bi	1.20E–09	6.28E–06	7.52E–15	3.23E–12	4.03E–08	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Bi	7.18E–08	6.28E–06	4.51E–13	1.94E–10	2.42E–06	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	1.89E–10	^{213}Bi	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Bi	2.88E–10	6.28E–06	1.81E–15	7.75E–13	9.68E–09	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	7.41E–08	^{14}C	[e]		2.21E–10	9.48E–08	1.69E–07	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	8.00E–16	^{109}Cd	2.38E–17	6.28E–06	1.49E–22	6.41E–20	8.01E–16	0.00%

Table A-1. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{113m}Cd	5.78E–05	1.8	1.04E–04	1.95E–03	^{113m}Cd	5.78E–05	6.28E–06	3.63E–10	1.56E–07	1.95E–03	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	2.46E–08	^{142}Ce	7.31E–10	6.28E–06	4.59E–15	1.97E–12	2.46E–08	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Ce	2.33E–10	6.28E–06	1.46E–15	6.27E–13	7.83E–09	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	2.43E–14	^{249}Cf	7.21E–16	6.28E–06	4.52E–21	1.94E–18	2.43E–14	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	1.26E–14	^{250}Cf	3.73E–16	6.28E–06	2.34E–21	1.00E–18	1.26E–14	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	3.85E–16	^{251}Cf	1.14E–17	6.28E–06	7.18E–23	3.08E–20	3.85E–16	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	1.63E–17	^{252}Cf	4.84E–19	6.28E–06	3.04E–24	1.30E–21	1.63E–17	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	1.99E–05	^{242}Cm	5.91E–07	6.28E–06	3.71E–12	1.59E–09	1.99E–05	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	4.33E–06	^{243}Cm	1.29E–07	6.28E–06	8.07E–13	3.46E–10	4.33E–06	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	2.37E–04	^{244}Cm	[e]		2.88E–12	1.24E–09	2.37E–04	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	5.66E–08	^{245}Cm	1.68E–09	6.28E–06	1.06E–14	4.53E–12	5.66E–08	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	3.71E–09	^{246}Cm	1.10E–10	6.28E–06	6.92E–16	2.97E–13	3.71E–09	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	4.11E–15	^{247}Cm	1.22E–16	6.28E–06	7.66E–22	3.29E–19	4.11E–15	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	4.34E–15	^{248}Cm	1.29E–16	6.28E–06	8.09E–22	3.47E–19	4.34E–15	0.00%
^{60}Co	[d]		5.02E–05	9.39E–04	^{60}Co	[e]		7.96E–10	3.41E–07	9.39E–04	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.02E–03	^{134}Cs	3.03E–05	6.28E–06	1.90E–10	8.17E–08	1.02E–03	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	4.84E–04	^{135}Cs	1.44E–05	6.28E–06	9.03E–11	3.87E–08	4.84E–04	0.00%
^{137}Cs	[d]		9.20E–01	1.72E+01	^{137}Cs	[e]		6.28E–06	2.69E–03	1.72E+01	47.48%
^{150}Eu	2.96E–10	1.8	5.34E–10	9.98E–09	^{150}Eu	2.96E–10	6.28E–06	1.86E–15	7.99E–13	9.98E–09	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	1.32E–03	^{152}Eu	3.92E–05	6.28E–06	2.46E–10	1.06E–07	1.32E–03	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	5.88E–02	^{154}Eu	[e]		3.57E–10	1.53E–07	5.88E–02	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	1.59E–02	^{155}Eu	4.74E–04	6.28E–06	2.97E–09	1.28E–06	1.59E–02	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	1.64E–02	^{55}Fe	4.88E–04	6.28E–06	3.07E–09	1.31E–06	1.64E–02	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	1.89E–10	^{221}Fr	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	5.55E–10	^{223}Fr	1.65E–11	6.28E–06	1.04E–16	4.44E–14	5.55E–10	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	1.20E–15	^{152}Gd	3.58E–17	6.28E–06	2.25E–22	9.64E–20	1.20E–15	0.00%

Table A-1. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{153}Gd	4.15E–19	1.8	7.48E–19	1.40E–17	^{153}Gd	4.15E–19	6.28E–06	2.61E–24	1.12E–21	1.40E–17	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.08E–02	^{3}H	[e]		2.48E–09	1.06E–06	1.08E–02	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	3.79E–08	^{166m}Ho	1.13E–09	6.28E–06	7.07E–15	3.03E–12	3.79E–08	0.00%
^{129}I	[d]		6.24E–07	1.17E–05	^{129}I	[e]		1.27E–10	5.45E–08	1.17E–05	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	9.26E–15	^{115}In	2.75E–16	6.28E–06	1.73E–21	7.41E–19	9.26E–15	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	1.58E–13	^{138}La	4.68E–15	6.28E–06	2.94E–20	1.26E–17	1.58E–13	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	2.94E–03	^{93m}Nb	8.74E–05	6.28E–06	5.49E–10	2.35E–07	2.94E–03	0.01%
^{94}Nb	[d]		1.66E–04	3.10E–03	^{94}Nb	[e]		4.28E–10	1.84E–07	3.10E–03	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	1.33E–12	^{144}Nd	3.96E–14	6.28E–06	2.49E–19	1.07E–16	1.33E–12	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	3.79E–04	^{59}Ni	1.13E–05	6.28E–06	7.08E–11	3.04E–08	3.79E–04	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	4.32E–02	^{63}Ni	[e]		3.70E–07	1.59E–04	4.33E–02	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	7.42E–09	^{236}Np	2.21E–10	6.28E–06	1.38E–15	5.94E–13	7.42E–09	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	7.10E–04	^{237}Np	[e]		1.86E–11	7.98E–09	7.10E–04	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	1.21E–07	^{238}Np	3.58E–09	6.28E–06	2.25E–14	9.66E–12	1.21E–07	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	3.31E–05	^{239}Np	9.83E–07	6.28E–06	6.17E–12	2.65E–09	3.31E–05	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	9.88E–13	^{240m}Np	2.94E–14	6.28E–06	1.84E–19	7.91E–17	9.88E–13	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	7.14E–08	^{231}Pa	2.12E–09	6.28E–06	1.33E–14	5.71E–12	7.14E–08	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	7.10E–04	^{233}Pa	2.11E–05	6.28E–06	1.33E–10	5.69E–08	7.10E–04	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	3.57E–08	^{234}Pa	1.06E–09	6.28E–06	6.67E–15	2.86E–12	3.57E–08	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	2.75E–05	^{234m}Pa	8.17E–07	6.28E–06	5.13E–12	2.20E–09	2.75E–05	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	1.89E–10	^{209}Pb	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Pb	1.11E–10	6.28E–06	7.00E–16	3.00E–13	3.75E–09	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Pb	1.20E–09	6.28E–06	7.52E–15	3.23E–12	4.03E–08	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Pb	7.18E–08	6.28E–06	4.51E–13	1.94E–10	2.42E–06	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Pb	2.88E–10	6.28E–06	1.81E–15	7.75E–13	9.68E–09	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	1.37E–05	^{107}Pd	4.06E–07	6.28E–06	2.55E–12	1.09E–09	1.37E–05	0.00%

Table A-1. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{146}Pm	4.00E–07	1.8	7.20E–07	1.35E–05	^{146}Pm	4.00E–07	6.28E–06	2.51E–12	1.08E–09	1.35E–05	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	1.31E–02	^{147}Pm	3.88E–04	6.28E–06	2.44E–09	1.05E–06	1.31E–02	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	3.62E–09	^{210}Po	1.08E–10	6.28E–06	6.76E–16	2.90E–13	3.62E–09	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	1.13E–10	^{211}Po	3.35E–12	6.28E–06	2.11E–17	9.03E–15	1.13E–10	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	1.55E–06	^{212}Po	4.60E–08	6.28E–06	2.89E–13	1.24E–10	1.55E–06	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	1.85E–10	^{213}Po	5.49E–12	6.28E–06	3.45E–17	1.48E–14	1.85E–10	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	9.68E–09	^{214}Po	2.87E–10	6.28E–06	1.81E–15	7.74E–13	9.68E–09	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	4.03E–08	^{215}Po	1.20E–09	6.28E–06	7.52E–15	3.23E–12	4.03E–08	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	2.42E–06	^{216}Po	7.18E–08	6.28E–06	4.51E–13	1.94E–10	2.42E–06	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	9.68E–09	^{218}Po	2.88E–10	6.28E–06	1.81E–15	7.75E–13	9.68E–09	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Pr	2.33E–10	6.28E–06	1.46E–15	6.27E–13	7.83E–09	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	9.40E–11	^{144m}Pr	2.79E–12	6.28E–06	1.75E–17	7.52E–15	9.40E–11	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	2.20E–07	^{236}Pu	6.52E–09	6.28E–06	4.10E–14	1.76E–11	2.20E–07	0.00%
^{238}Pu	[d]		9.23E–03	1.73E–01	^{238}Pu	[e]		3.79E–10	1.63E–07	1.73E–01	0.48%
^{239}Pu	[d]		2.75E–03	5.14E–02	^{239}Pu	[e]		2.90E–11	1.24E–08	5.14E–02	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	2.04E–02	^{240}Pu	6.06E–04	6.28E–06	3.81E–09	1.63E–06	2.04E–02	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	2.95E–01	^{241}Pu	[e]		4.40E–08	1.89E–05	2.95E–01	0.81%
^{242}Pu	4.43E–07	1.8	7.98E–07	1.49E–05	^{242}Pu	4.43E–07	6.28E–06	2.78E–12	1.19E–09	1.49E–05	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	4.11E–15	^{243}Pu	1.22E–16	6.28E–06	7.66E–22	3.29E–19	4.11E–15	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	9.90E–13	^{244}Pu	2.94E–14	6.28E–06	1.85E–19	7.92E–17	9.90E–13	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	4.03E–08	^{223}Ra	1.20E–09	6.28E–06	7.52E–15	3.23E–12	4.03E–08	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	2.42E–06	^{224}Ra	7.18E–08	6.28E–06	4.51E–13	1.94E–10	2.42E–06	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ra	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	9.68E–09	^{226}Ra	2.88E–10	6.28E–06	1.81E–15	7.75E–13	9.68E–09	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ra	1.73E–14	6.28E–06	1.08E–19	4.65E–17	5.81E–13	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	2.41E–08	^{87}Rb	7.15E–10	6.28E–06	4.49E–15	1.93E–12	2.41E–08	0.00%

Table A-1. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{102}Rh	2.46E–09	1.8	4.44E–09	8.29E–08	^{102}Rh	2.46E–09	6.28E–06	1.55E–14	6.64E–12	8.30E–08	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Rh	7.69E–09	6.28E–06	4.83E–14	2.07E–11	2.59E–07	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	4.03E–08	^{219}Rn	1.20E–09	6.28E–06	7.52E–15	3.23E–12	4.03E–08	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	2.42E–06	^{220}Rn	7.18E–08	6.28E–06	4.51E–13	1.94E–10	2.42E–06	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	9.68E–09	^{222}Rn	2.88E–10	6.28E–06	1.81E–15	7.75E–13	9.68E–09	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Ru	7.69E–09	6.28E–06	4.83E–14	2.07E–11	2.59E–07	0.00%
^{125}Sb	[d]		5.55E–04	1.04E–02	^{125}Sb	[e]		2.55E–10	1.09E–07	1.04E–02	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	4.76E–05	^{126}Sb	1.41E–06	6.28E–06	8.88E–12	3.81E–09	4.76E–05	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	3.40E–04	^{126m}Sb	1.01E–05	6.28E–06	6.35E–11	2.72E–08	3.40E–04	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	3.61E–04	^{79}Se	1.07E–05	6.28E–06	6.74E–11	2.89E–08	3.61E–04	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	1.56E–10	^{146}Sm	4.65E–12	6.28E–06	2.92E–17	1.25E–14	1.56E–10	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	6.10E–09	^{147}Sm	1.81E–10	6.28E–06	1.14E–15	4.88E–13	6.10E–09	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	3.13E–14	^{148}Sm	9.30E–16	6.28E–06	5.84E–21	2.51E–18	3.13E–14	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	2.78E–15	^{149}Sm	8.26E–17	6.28E–06	5.19E–22	2.22E–19	2.78E–15	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	2.59E–01	^{151}Sm	7.71E–03	6.28E–06	4.84E–08	2.08E–05	2.59E–01	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	3.21E–13	^{119m}Sn	9.55E–15	6.28E–06	6.00E–20	2.57E–17	3.21E–13	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	1.93E–03	^{121m}Sn	5.74E–05	6.28E–06	3.60E–10	1.55E–07	1.93E–03	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	3.40E–04	^{126}Sn	1.01E–05	6.28E–06	6.35E–11	2.72E–08	3.40E–04	0.00%
^{90}Sr	[d]		1.87E–02	3.50E–01	^{90}Sr	[e]		2.78E–04	1.19E–01	4.69E–01	1.29%
^{98}Tc	6.34E–11	1.8	1.14E–10	2.14E–09	^{98}Tc	6.34E–11	6.28E–06	3.98E–16	1.71E–13	2.14E–09	0.00%
^{99}Tc	[d]		6.17E–04	1.15E–02	^{99}Tc	[e]		2.12E–08	9.09E–06	1.15E–02	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	1.72E–15	^{123}Te	5.12E–17	6.28E–06	3.22E–22	1.38E–19	1.72E–15	0.00%
^{125m}Te	1.43E–05		1.36E–04	2.54E–03	^{125m}Te	1.43E–05	6.28E–06	9.00E–11	3.86E–08	2.54E–03	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	3.98E–08	^{227}Th	1.18E–09	6.28E–06	7.42E–15	3.18E–12	3.98E–08	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	2.41E–06	^{228}Th	7.16E–08	6.28E–06	4.50E–13	1.93E–10	2.41E–06	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	1.89E–10	^{229}Th	5.61E–12	6.28E–06	3.52E–17	1.51E–14	1.89E–10	0.00%

Table A-1. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{230}Th	2.35E–08	1.8	4.24E–08	7.92E–07	^{230}Th	2.35E–08	6.28E–06	1.48E–13	6.34E–11	7.92E–07	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	2.67E–05	^{231}Th	7.93E–07	6.28E–06	4.98E–12	2.14E–09	2.67E–05	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	6.23E–13	^{232}Th	1.85E–14	6.28E–06	1.16E–19	4.98E–17	6.23E–13	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	2.75E–05	^{234}Th	8.17E–07	6.28E–06	5.13E–12	2.20E–09	2.75E–05	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	4.02E–08	^{207}Tl	1.19E–09	6.28E–06	7.50E–15	3.22E–12	4.02E–08	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	8.69E–07	^{208}Tl	2.58E–08	6.28E–06	1.62E–13	6.95E–11	8.69E–07	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	4.08E–12	^{209}Tl	1.21E–13	6.28E–06	7.61E–19	3.26E–16	4.08E–12	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	1.61E–14	^{171}Tm	4.80E–16	6.28E–06	3.01E–21	1.29E–18	1.61E–14	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	2.33E–06	^{232}U	6.91E–08	6.28E–06	4.34E–13	1.86E–10	2.33E–06	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	5.97E–08	^{233}U	1.77E–09	6.28E–06	1.11E–14	4.78E–12	5.97E–08	0.00%
^{234}U	[d]		2.98E–06	5.57E–05	^{234}U	[e]		7.28E–12	3.12E–09	5.57E–05	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	2.67E–05	^{235}U	7.93E–07	6.28E–06	4.98E–12	2.14E–09	2.67E–05	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	6.22E–05	^{236}U	1.85E–06	6.28E–06	1.16E–11	4.98E–09	6.22E–05	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	7.23E–06	^{237}U	2.15E–07	6.28E–06	1.35E–12	5.78E–10	7.23E–06	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	2.75E–05	^{238}U	8.17E–07	6.28E–06	5.13E–12	2.20E–09	2.75E–05	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	9.88E–13	^{240}U	2.94E–14	6.28E–06	1.84E–19	7.91E–17	9.88E–13	0.00%
^{90}Y	[d]		1.87E–02	3.50E–01	^{90}Y	8.88E–01	6.28E–06	5.579E–06	2.39E–03	3.52E–01	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	3.56E–03	^{93}Zr	1.06E–04	6.28E–06	6.64E–10	2.85E–07	3.56E–03	0.01%
			Total	36.1				Total	0.2	36.4	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-103.

Table A-2. Post-decontamination estimated inventory for Tank WM-104.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ac	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	4.02E–08	^{227}Ac	1.20E–09	2.37E–05	2.83E–14	1.22E–11	4.02E–08	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ac	1.73E–14	2.37E–05	4.09E–19	1.76E–16	5.82E–13	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	9.59E–12	^{108}Ag	2.85E–13	2.37E–05	6.75E–18	2.90E–15	9.59E–12	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.08E–10	^{108m}Ag	3.20E–12	2.37E–05	7.59E–17	3.26E–14	1.08E–10	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	8.00E–16	^{109m}Ag	2.38E–17	2.37E–05	5.64E–22	2.42E–19	8.01E–16	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	3.01E–17	^{110}Ag	8.93E–19	2.37E–05	2.12E–23	9.08E–21	3.01E–17	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	2.26E–15	^{110m}Ag	6.71E–17	2.37E–05	1.59E–21	6.83E–19	2.26E–15	0.00%
^{241}Am	[d]		3.40E–04	6.36E–03	^{241}Am	[e]		5.86E–09	2.51E–06	6.36E–03	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	2.40E–05	^{242}Am	7.13E–07	2.37E–05	1.69E–11	7.25E–09	2.40E–05	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	2.41E–05	^{242m}Am	7.17E–07	2.37E–05	1.70E–11	7.29E–09	2.41E–05	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	3.31E–05	^{243}Am	9.83E–07	2.37E–05	2.33E–11	1.00E–08	3.31E–05	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	1.89E–10	^{217}At	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{137m}Ba	[d]		9.20E–01	1.72E+01	^{137m}Ba	[e]		2.37E–05	1.02E–02	1.72E+01	47.62%
^{10}Be	7.56E–11	1.8	1.36E–10	2.55E–09	^{10}Be	7.56E–11	2.37E–05	1.79E–15	7.69E–13	2.55E–09	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Bi	1.11E–10	2.37E–05	2.64E–15	1.13E–12	3.75E–09	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	1.82E–22	^{210m}Bi	5.41E–24	2.37E–05	1.28E–28	5.50E–26	1.82E–22	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Bi	1.20E–09	2.37E–05	2.84E–14	1.22E–11	4.03E–08	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Bi	7.18E–08	2.37E–05	1.70E–12	7.30E–10	2.42E–06	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	1.89E–10	^{213}Bi	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Bi	2.88E–10	2.37E–05	6.81E–15	2.92E–12	9.68E–09	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	7.41E–08	^{14}C	[e]		8.29E–11	3.56E–08	1.10E–07	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	8.00E–16	^{109}Cd	2.38E–17	2.37E–05	5.64E–22	2.42E–19	8.01E–16	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	1.95E–03	^{113m}Cd	5.78E–05	2.37E–05	1.37E–09	5.88E–07	1.95E–03	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	2.46E–08	^{142}Ce	7.31E–10	2.37E–05	1.73E–14	7.44E–12	2.46E–08	0.00%

Table A-2. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{144}Ce	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Ce	2.33E–10	2.37E–05	5.51E–15	2.37E–12	7.83E–09	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	2.43E–14	^{249}Cf	7.21E–16	2.37E–05	1.71E–20	7.33E–18	2.43E–14	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	1.26E–14	^{250}Cf	3.73E–16	2.37E–05	8.84E–21	3.79E–18	1.26E–14	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	3.85E–16	^{251}Cf	1.14E–17	2.37E–05	2.71E–22	1.16E–19	3.85E–16	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	1.63E–17	^{252}Cf	4.84E–19	2.37E–05	1.15E–23	4.92E–21	1.63E–17	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	1.99E–05	^{242}Cm	5.91E–07	2.37E–05	1.40E–11	6.01E–09	1.99E–05	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	4.33E–06	^{243}Cm	1.29E–07	2.37E–05	3.05E–12	1.31E–09	4.33E–06	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	2.37E–04	^{244}Cm	[e]		1.31E–21	5.62E–19	2.37E–04	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	5.66E–08	^{245}Cm	1.68E–09	2.37E–05	3.98E–14	1.71E–11	5.66E–08	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	3.71E–09	^{246}Cm	1.10E–10	2.37E–05	2.61E–15	1.12E–12	3.71E–09	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	4.11E–15	^{247}Cm	1.22E–16	2.37E–05	2.89E–21	1.24E–18	4.11E–15	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	4.34E–15	^{248}Cm	1.29E–16	2.37E–05	3.05E–21	1.31E–18	4.34E–15	0.00%
^{60}Co	[d]		5.02E–05	9.39E–04	^{60}Co	[e]		6.98E–09	2.99E–06	9.42E–04	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.02E–03	^{134}Cs	3.03E–05	2.37E–05	7.19E–10	3.08E–07	1.02E–03	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	4.84E–04	^{135}Cs	1.44E–05	2.37E–05	3.41E–10	1.46E–07	4.84E–04	0.00%
^{137}Cs	[d]		9.20E–01	1.72E+01	^{137}Cs	[e]		2.37E–05	1.02E–02	1.72E+01	47.62%
^{150}Eu	2.96E–10	1.8	5.34E–10	9.98E–09	^{150}Eu	2.96E–10	2.37E–05	7.03E–15	3.01E–12	9.98E–09	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	1.32E–03	^{152}Eu	3.92E–05	2.37E–05	9.29E–10	3.98E–07	1.32E–03	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	5.88E–02	^{154}Eu	[e]		8.79E–09	3.77E–06	5.88E–02	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	1.59E–02	^{155}Eu	4.74E–04	2.37E–05	1.12E–08	4.82E–06	1.59E–02	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	1.64E–02	^{55}Fe	4.88E–04	2.37E–05	1.16E–08	4.96E–06	1.64E–02	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	1.89E–10	^{221}Fr	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	5.55E–10	^{223}Fr	1.65E–11	2.37E–05	3.91E–16	1.68E–13	5.55E–10	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	1.20E–15	^{152}Gd	3.58E–17	2.37E–05	8.48E–22	3.64E–19	1.20E–15	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	1.40E–17	^{153}Gd	4.15E–19	2.37E–05	9.84E–24	4.22E–21	1.40E–17	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.08E–02	^{3}H	[e]		1.41E–08	6.05E–06	1.08E–02	0.03%

Table A-2. (continued).

	Solids				Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{166m}Ho	1.13E–09	1.8	2.03E–09	3.79E–08	^{166m}Ho	1.13E–09	2.37E–05	2.67E–14	1.14E–11	3.79E–08	0.00%
^{129}I	[d]		6.24E–07	1.17E–05	^{129}I	[e]		1.36E–10	5.83E–08	1.17E–05	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	9.26E–15	^{115}In	2.75E–16	2.37E–05	6.52E–21	2.80E–18	9.26E–15	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	1.58E–13	^{138}La	4.68E–15	2.37E–05	1.11E–19	4.76E–17	1.58E–13	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	2.94E–03	^{93m}Nb	8.74E–05	2.37E–05	2.07E–09	8.88E–07	2.94E–03	0.01%
^{94}Nb	[d]		1.66E–04	3.10E–03	^{94}Nb	[e]		1.02E–08	4.38E–06	3.11E–03	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	1.33E–12	^{144}Nd	3.96E–14	2.37E–05	9.40E–19	4.03E–16	1.33E–12	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	3.79E–04	^{59}Ni	1.13E–05	2.37E–05	2.67E–10	1.15E–07	3.79E–04	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	4.32E–02	^{63}Ni	[e]		8.79E–09	3.77E–06	4.32E–02	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	7.42E–09	^{236}Np	2.21E–10	2.37E–05	5.23E–15	2.24E–12	7.42E–09	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	7.10E–04	^{237}Np	[e]		1.06E–10	4.55E–08	7.10E–04	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	1.21E–07	^{238}Np	3.58E–09	2.37E–05	8.50E–14	3.64E–11	1.21E–07	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	3.31E–05	^{239}Np	9.83E–07	2.37E–05	2.33E–11	1.00E–08	3.31E–05	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	9.88E–13	^{240m}Np	2.94E–14	2.37E–05	6.96E–19	2.99E–16	9.89E–13	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	7.14E–08	^{231}Pa	2.12E–09	2.37E–05	5.03E–14	2.16E–11	7.14E–08	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	7.10E–04	^{233}Pa	2.11E–05	2.37E–05	5.00E–10	2.15E–07	7.11E–04	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	3.57E–08	^{234}Pa	1.06E–09	2.37E–05	2.52E–14	1.08E–11	3.57E–08	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	2.75E–05	^{234m}Pa	8.17E–07	2.37E–05	1.94E–11	8.30E–09	2.75E–05	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	1.89E–10	^{209}Pb	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Pb	1.11E–10	2.37E–05	2.64E–15	1.13E–12	3.75E–09	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Pb	1.20E–09	2.37E–05	2.84E–14	1.22E–11	4.03E–08	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Pb	7.18E–08	2.37E–05	1.70E–12	7.30E–10	2.42E–06	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Pb	2.88E–10	2.37E–05	6.81E–15	2.92E–12	9.68E–09	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	1.37E–05	^{107}Pd	4.06E–07	2.37E–05	9.62E–12	4.13E–09	1.37E–05	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	1.35E–05	^{146}Pm	4.00E–07	2.37E–05	9.48E–12	4.07E–09	1.35E–05	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	1.31E–02	^{147}Pm	3.88E–04	2.37E–05	9.20E–09	3.95E–06	1.31E–02	0.04%

Table A-2. (continued).

	Solids				Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{210}Po	1.08E–10	1.8	1.94E–10	3.62E–09	^{210}Po	1.08E–10	2.37E–05	2.55E–15	1.09E–12	3.62E–09	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	1.13E–10	^{211}Po	3.35E–12	2.37E–05	7.95E–17	3.41E–14	1.13E–10	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	1.55E–06	^{212}Po	4.60E–08	2.37E–05	1.09E–12	4.68E–10	1.55E–06	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	1.85E–10	^{213}Po	5.49E–12	2.37E–05	1.30E–16	5.58E–14	1.85E–10	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	9.68E–09	^{214}Po	2.87E–10	2.37E–05	6.81E–15	2.92E–12	9.68E–09	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	4.03E–08	^{215}Po	1.20E–09	2.37E–05	2.84E–14	1.22E–11	4.03E–08	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	2.42E–06	^{216}Po	7.18E–08	2.37E–05	1.70E–12	7.30E–10	2.42E–06	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	9.68E–09	^{218}Po	2.88E–10	2.37E–05	6.82E–15	2.92E–12	9.68E–09	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Pr	2.33E–10	2.37E–05	5.51E–15	2.37E–12	7.83E–09	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	9.40E–11	^{144m}Pr	2.79E–12	2.37E–05	6.62E–17	2.84E–14	9.40E–11	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	2.20E–07	^{236}Pu	6.52E–09	2.37E–05	1.55E–13	6.63E–11	2.20E–07	0.00%
^{238}Pu	[d]		9.23E–03	1.73E–01	^{238}Pu	[e]		2.30E–07	9.87E–05	1.73E–01	0.48%
^{239}Pu	[d]		2.75E–03	5.14E–02	^{239}Pu	[e]		2.87E–08	1.23E–05	5.14E–02	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	2.04E–02	^{240}Pu	6.06E–04	2.37E–05	1.44E–08	6.16E–06	2.04E–02	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	2.95E–01	^{241}Pu	[e]		1.10E–07	4.72E–05	2.95E–01	0.81%
^{242}Pu	4.43E–07	1.8	7.98E–07	1.49E–05	^{242}Pu	4.43E–07	2.37E–05	1.05E–11	4.51E–09	1.49E–05	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	4.11E–15	^{243}Pu	1.22E–16	2.37E–05	2.89E–21	1.24E–18	4.11E–15	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	9.90E–13	^{244}Pu	2.94E–14	2.37E–05	6.97E–19	2.99E–16	9.90E–13	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	4.03E–08	^{223}Ra	1.20E–09	2.37E–05	2.84E–14	1.22E–11	4.03E–08	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	2.42E–06	^{224}Ra	7.18E–08	2.37E–05	1.70E–12	7.30E–10	2.42E–06	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ra	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	9.68E–09	^{226}Ra	2.88E–10	2.37E–05	6.82E–15	2.92E–12	9.68E–09	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ra	1.73E–14	2.37E–05	4.09E–19	1.76E–16	5.82E–13	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	2.41E–08	^{87}Rb	7.15E–10	2.37E–05	1.69E–14	7.26E–12	2.41E–08	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	8.29E–08	^{102}Rh	2.46E–09	2.37E–05	5.84E–14	2.51E–11	8.30E–08	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Rh	7.69E–09	2.37E–05	1.82E–13	7.81E–11	2.59E–07	0.00%

Table A-2. (continued).

	Solids					Liquids				Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{219}Rn	1.20E–09	1.8	2.16E–09	4.03E–08	^{219}Rn	1.20E–09	2.37E–05	2.84E–14	1.22E–11	4.03E–08	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	2.42E–06	^{220}Rn	7.18E–08	2.37E–05	1.70E–12	7.30E–10	2.42E–06	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	9.68E–09	^{222}Rn	2.88E–10	2.37E–05	6.82E–15	2.92E–12	9.68E–09	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Ru	7.69E–09	2.37E–05	1.82E–13	7.81E–11	2.59E–07	0.00%
^{125}Sb	[d]		5.55E–04	1.04E–02	^{125}Sb	[e]		1.11E–08	4.76E–06	1.04E–02	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	4.76E–05	^{126}Sb	1.41E–06	2.37E–05	3.35E–11	1.44E–08	4.76E–05	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	3.40E–04	^{126m}Sb	1.01E–05	2.37E–05	2.39E–10	1.03E–07	3.40E–04	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	3.61E–04	^{79}Se	1.07E–05	2.37E–05	2.54E–10	1.09E–07	3.61E–04	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	1.56E–10	^{146}Sm	4.65E–12	2.37E–05	1.10E–16	4.72E–14	1.56E–10	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	6.10E–09	^{147}Sm	1.81E–10	2.37E–05	4.30E–15	1.84E–12	6.10E–09	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	3.13E–14	^{148}Sm	9.30E–16	2.37E–05	2.20E–20	9.46E–18	3.13E–14	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	2.78E–15	^{149}Sm	8.26E–17	2.37E–05	1.96E–21	8.40E–19	2.78E–15	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	2.59E–01	^{151}Sm	7.71E–03	2.37E–05	1.83E–07	7.84E–05	2.60E–01	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	3.21E–13	^{119m}Sn	9.55E–15	2.37E–05	2.26E–19	9.71E–17	3.21E–13	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	1.93E–03	^{121m}Sn	5.74E–05	2.37E–05	1.36E–09	5.83E–07	1.93E–03	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	3.40E–04	^{126}Sn	1.01E–05	2.37E–05	2.39E–10	1.03E–07	3.40E–04	0.00%
^{90}Sr	[d]		1.87E–02	3.50E–01	^{90}Sr	[e]		1.47E–05	6.31E–03	3.56E–01	0.98%
^{98}Tc	6.34E–11	1.8	1.14E–10	2.14E–09	^{98}Tc	6.34E–11	2.37E–05	1.50E–15	6.45E–13	2.14E–09	0.00%
^{99}Tc	[d]		6.17E–04	1.15E–02	^{99}Tc	[e]		6.95E–09	2.98E–06	1.15E–02	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	1.72E–15	^{123}Te	5.12E–17	2.37E–05	1.21E–21	5.21E–19	1.72E–15	0.00%
^{125m}Te	1.43E–05		1.36E–04	2.54E–03	^{125m}Te	1.43E–05	2.37E–05	3.40E–10	1.46E–07	2.54E–03	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	3.98E–08	^{227}Th	1.18E–09	2.37E–05	2.80E–14	1.20E–11	3.98E–08	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	2.41E–06	^{228}Th	7.16E–08	2.37E–05	1.70E–12	7.28E–10	2.41E–06	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	1.89E–10	^{229}Th	5.61E–12	2.37E–05	1.33E–16	5.70E–14	1.89E–10	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	7.92E–07	^{230}Th	2.35E–08	2.37E–05	5.58E–13	2.39E–10	7.92E–07	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	2.67E–05	^{231}Th	7.93E–07	2.37E–05	1.88E–11	8.06E–09	2.67E–05	0.00%

Table A-2. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{232}Th	1.85E–14	1.8	3.33E–14	6.23E–13	^{232}Th	1.85E–14	2.37E–05	4.38E–19	1.88E–16	6.23E–13	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	2.75E–05	^{234}Th	8.17E–07	2.37E–05	1.94E–11	8.30E–09	2.75E–05	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	4.02E–08	^{207}Tl	1.19E–09	2.37E–05	2.83E–14	1.21E–11	4.02E–08	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	8.69E–07	^{208}Tl	2.58E–08	2.37E–05	6.12E–13	2.62E–10	8.69E–07	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	4.08E–12	^{209}Tl	1.21E–13	2.37E–05	2.87E–18	1.23E–15	4.08E–12	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	1.61E–14	^{171}Tm	4.80E–16	2.37E–05	1.14E–20	4.88E–18	1.61E–14	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	2.33E–06	^{232}U	6.91E–08	2.37E–05	1.64E–12	7.03E–10	2.33E–06	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	5.97E–08	^{233}U	1.77E–09	2.37E–05	4.21E–14	1.80E–11	5.98E–08	0.00%
^{234}U	[d]		2.98E–06	5.57E–05	^{234}U	[e]		1.38E–09	5.92E–07	5.63E–05	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	2.67E–05	^{235}U	[e]		3.40E–10	1.46E–07	2.68E–05	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	6.22E–05	^{236}U	1.85E–06	2.37E–05	4.38E–11	1.88E–08	6.23E–05	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	7.23E–06	^{237}U	2.15E–07	2.37E–05	5.09E–12	2.18E–09	7.23E–06	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	2.75E–05	^{238}U	[e]		1.11E–10	4.76E–08	2.75E–05	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	9.88E–13	^{240}U	2.94E–14	2.37E–05	6.96E–19	2.99E–16	9.89E–13	0.00%
^{90}Y	[d]		1.87E–02	3.50E–01	^{90}Y	8.88E–01	2.37E–05	2.11E–05	9.03E–03	3.59E–01	0.99%
^{93}Zr	1.06E–04	1.8	1.90E–04	3.56E–03	^{93}Zr	1.06E–04	2.37E–05	2.51E–09	1.07E–06	3.56E–03	0.01%
			Total	36.12				Total	0.03	36.15	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-104.

Table A-3. Post-decontamination estimated inventory for Tank WM-105.

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ac	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	4.02E–08	^{227}Ac	1.20E–09	2.05E–05	2.45E–14	1.05E–11	4.02E–08	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ac	1.73E–14	2.05E–05	3.54E–19	1.52E–16	5.82E–13	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	9.59E–12	^{108}Ag	2.85E–13	2.05E–05	5.84E–18	2.51E–15	9.59E–12	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.08E–10	^{108m}Ag	3.20E–12	2.05E–05	6.56E–17	2.82E–14	1.08E–10	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	8.00E–16	^{109m}Ag	2.38E–17	2.05E–05	4.88E–22	2.09E–19	8.01E–16	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	3.01E–17	^{110}Ag	8.93E–19	2.05E–05	1.83E–23	7.85E–21	3.01E–17	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	2.26E–15	^{110m}Ag	6.71E–17	2.05E–05	1.38E–21	5.90E–19	2.26E–15	0.00%
^{241}Am	[d]		3.40E–04	6.36E–03	^{241}Am	[e]		2.82E–09	1.21E–06	6.36E–03	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	2.40E–05	^{242}Am	7.13E–07	2.05E–05	1.46E–11	6.27E–09	2.40E–05	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	2.41E–05	^{242m}Am	7.17E–07	2.05E–05	1.47E–11	6.31E–09	2.41E–05	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	3.31E–05	^{243}Am	9.83E–07	2.05E–05	2.02E–11	8.65E–09	3.31E–05	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	1.89E–10	^{217}At	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{137m}Ba	[d]		9.20E–01	1.72E+01	^{137m}Ba	[e]		2.05E–05	8.79E–03	1.72E+01	47.62%
^{10}Be	7.56E–11	1.8	1.36E–10	2.55E–09	^{10}Be	7.56E–11	2.05E–05	1.55E–15	6.65E–13	2.55E–09	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Bi	1.11E–10	2.05E–05	2.28E–15	9.80E–13	3.75E–09	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	1.82E–22	^{210m}Bi	5.41E–24	2.05E–05	1.11E–28	4.76E–26	1.82E–22	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Bi	1.20E–09	2.05E–05	2.46E–14	1.05E–11	4.03E–08	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Bi	7.18E–08	2.05E–05	1.47E–12	6.32E–10	2.42E–06	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	1.89E–10	^{213}Bi	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Bi	2.88E–10	2.05E–05	5.89E–15	2.53E–12	9.68E–09	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	7.41E–08	^{14}C	[e]		3.19E–11	1.37E–08	8.77E–08	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	8.00E–16	^{109}Cd	2.38E–17	2.05E–05	4.88E–22	2.09E–19	8.01E–16	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	1.95E–03	^{113m}Cd	5.78E–05	2.05E–05	1.18E–09	5.08E–07	1.95E–03	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	2.46E–08	^{142}Ce	7.31E–10	2.05E–05	1.50E–14	6.43E–12	2.46E–08	0.00%

Table A-3. (continued).

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{144}Ce	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Ce	2.33E–10	2.05E–05	4.77E–15	2.05E–12	7.83E–09	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	2.43E–14	^{249}Cf	7.21E–16	2.05E–05	1.48E–20	6.34E–18	2.43E–14	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	1.26E–14	^{250}Cf	3.73E–16	2.05E–05	7.64E–21	3.28E–18	1.26E–14	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	3.85E–16	^{251}Cf	1.14E–17	2.05E–05	2.34E–22	1.00E–19	3.85E–16	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	1.63E–17	^{252}Cf	4.84E–19	2.05E–05	9.93E–24	4.26E–21	1.63E–17	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	1.99E–05	^{242}Cm	5.91E–07	2.05E–05	1.21E–11	5.20E–09	1.99E–05	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	4.33E–06	^{243}Cm	1.29E–07	2.05E–05	2.64E–12	1.13E–09	4.33E–06	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	2.37E–04	^{244}Cm	[e]		6.83E–11	2.93E–08	2.37E–04	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	5.66E–08	^{245}Cm	1.68E–09	2.05E–05	3.44E–14	1.48E–11	5.66E–08	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	3.71E–09	^{246}Cm	1.10E–10	2.05E–05	2.26E–15	9.69E–13	3.71E–09	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	4.11E–15	^{247}Cm	1.22E–16	2.05E–05	2.50E–21	1.07E–18	4.11E–15	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	4.34E–15	^{248}Cm	1.29E–16	2.05E–05	2.64E–21	1.13E–18	4.34E–15	0.00%
^{60}Co	[d]		5.02E–05	9.39E–04	^{60}Co	[e]		2.35E–09	1.01E–06	9.40E–04	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.02E–03	^{134}Cs	3.03E–05	2.05E–05	6.22E–10	2.67E–07	1.02E–03	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	4.84E–04	^{135}Cs	1.44E–05	2.05E–05	2.95E–10	1.26E–07	4.84E–04	0.00%
^{137}Cs	[d]		9.20E–01	1.72E+01	^{137}Cs	[e]		2.05E–05	8.79E–03	1.72E+01	47.62%
^{150}Eu	2.96E–10	1.8	5.34E–10	9.98E–09	^{150}Eu	2.96E–10	2.05E–05	6.08E–15	2.61E–12	9.98E–09	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	1.32E–03	^{152}Eu	3.92E–05	2.05E–05	8.03E–10	3.45E–07	1.32E–03	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	5.88E–02	^{154}Eu	[e]		3.55E–09	1.52E–06	5.88E–02	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	1.59E–02	^{155}Eu	4.74E–04	2.05E–05	9.71E–09	4.17E–06	1.59E–02	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	1.64E–02	^{55}Fe	4.88E–04	2.05E–05	1.00E–08	4.29E–06	1.64E–02	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	1.89E–10	^{221}Fr	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	5.55E–10	^{223}Fr	1.65E–11	2.05E–05	3.38E–16	1.45E–13	5.55E–10	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	1.20E–15	^{152}Gd	3.58E–17	2.05E–05	7.33E–22	3.15E–19	1.20E–15	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	1.40E–17	^{153}Gd	4.15E–19	2.05E–05	8.52E–24	3.65E–21	1.40E–17	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.08E–02	^{3}H	[e]		1.28E–08	5.49E–06	1.08E–02	0.03%

Table A-3. (continued).

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{166m}Ho	1.13E–09	1.8	2.03E–09	3.79E–08	^{166m}Ho	1.13E–09	2.05E–05	2.31E–14	9.90E–12	3.79E–08	0.00%
^{129}I	[d]		6.24E–07	1.17E–05	^{129}I	[e]		1.05E–10	4.50E–08	1.17E–05	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	9.26E–15	^{115}In	2.75E–16	2.05E–05	5.64E–21	2.42E–18	9.26E–15	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	1.58E–13	^{138}La	4.68E–15	2.05E–05	9.60E–20	4.12E–17	1.58E–13	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	2.94E–03	^{93m}Nb	8.74E–05	2.05E–05	1.79E–09	7.68E–07	2.94E–03	0.01%
^{94}Nb	[d]		1.66E–04	3.10E–03	^{94}Nb	3.62E–05	2.05E–05	7.42E–10	3.18E–07	3.10E–03	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	1.33E–12	^{144}Nd	3.96E–14	2.05E–05	8.13E–19	3.49E–16	1.33E–12	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	3.79E–04	^{59}Ni	1.13E–05	2.05E–05	2.31E–10	9.91E–08	3.79E–04	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	4.32E–02	^{63}Ni	[e]		8.00E–09	3.43E–06	4.32E–02	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	7.42E–09	^{236}Np	2.21E–10	2.05E–05	4.52E–15	1.94E–12	7.42E–09	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	7.10E–04	^{237}Np	[e]		2.54E–11	1.09E–08	7.10E–04	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	1.21E–07	^{238}Np	3.58E–09	2.05E–05	7.35E–14	3.15E–11	1.21E–07	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	3.31E–05	^{239}Np	9.83E–07	2.05E–05	2.02E–11	8.65E–09	3.31E–05	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	9.88E–13	^{240m}Np	2.94E–14	2.05E–05	6.02E–19	2.58E–16	9.89E–13	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	7.14E–08	^{231}Pa	2.12E–09	2.05E–05	4.35E–14	1.87E–11	7.14E–08	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	7.10E–04	^{233}Pa	2.11E–05	2.05E–05	4.33E–10	1.86E–07	7.10E–04	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	3.57E–08	^{234}Pa	1.06E–09	2.05E–05	2.18E–14	9.33E–12	3.57E–08	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	2.75E–05	^{234m}Pa	8.17E–07	2.05E–05	1.67E–11	7.18E–09	2.75E–05	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	1.89E–10	^{209}Pb	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Pb	1.11E–10	2.05E–05	2.28E–15	9.80E–13	3.75E–09	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Pb	1.20E–09	2.05E–05	2.46E–14	1.05E–11	4.03E–08	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Pb	7.18E–08	2.05E–05	1.47E–12	6.32E–10	2.42E–06	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Pb	2.88E–10	2.05E–05	5.89E–15	2.53E–12	9.68E–09	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	1.37E–05	^{107}Pd	4.06E–07	2.05E–05	8.32E–12	3.57E–09	1.37E–05	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	1.35E–05	^{146}Pm	4.00E–07	2.05E–05	8.20E–12	3.52E–09	1.35E–05	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	1.31E–02	^{147}Pm	3.88E–04	2.05E–05	7.96E–09	3.42E–06	1.31E–02	0.04%

Table A-3. (continued).

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{210}Po	1.08E–10	1.8	1.94E–10	3.62E–09	^{210}Po	1.08E–10	2.05E–05	2.21E–15	9.46E–13	3.62E–09	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	1.13E–10	^{211}Po	3.35E–12	2.05E–05	6.87E–17	2.95E–14	1.13E–10	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	1.55E–06	^{212}Po	4.60E–08	2.05E–05	9.44E–13	4.05E–10	1.55E–06	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	1.85E–10	^{213}Po	5.49E–12	2.05E–05	1.13E–16	4.83E–14	1.85E–10	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	9.68E–09	^{214}Po	2.87E–10	2.05E–05	5.89E–15	2.53E–12	9.68E–09	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	4.03E–08	^{215}Po	1.20E–09	2.05E–05	2.46E–14	1.05E–11	4.03E–08	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	2.42E–06	^{216}Po	7.18E–08	2.05E–05	1.47E–12	6.32E–10	2.42E–06	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	9.68E–09	^{218}Po	2.88E–10	2.05E–05	5.90E–15	2.53E–12	9.68E–09	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Pr	2.33E–10	2.05E–05	4.77E–15	2.05E–12	7.83E–09	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	9.40E–11	^{144m}Pr	2.79E–12	2.05E–05	5.72E–17	2.46E–14	9.40E–11	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	2.20E–07	^{236}Pu	6.52E–09	2.05E–05	1.34E–13	5.74E–11	2.20E–07	0.00%
^{238}Pu	[d]		9.23E–03	1.73E–01	^{238}Pu	[e]		2.43E–08	1.04E–05	1.73E–01	0.48%
^{239}Pu	[d]		2.75E–03	5.14E–02	^{239}Pu	[e]		2.57E–09	1.10E–06	5.14E–02	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	2.04E–02	^{240}Pu	6.06E–04	2.05E–05	1.24E–08	5.33E–06	2.04E–02	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	2.95E–01	^{241}Pu	[e]		1.58E–08	6.78E–06	2.95E–01	0.81%
^{242}Pu	4.43E–07	1.8	7.98E–07	1.49E–05	^{242}Pu	4.43E–07	2.05E–05	9.08E–12	3.90E–09	1.49E–05	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	4.11E–15	^{243}Pu	1.22E–16	2.05E–05	2.50E–21	1.07E–18	4.11E–15	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	9.90E–13	^{244}Pu	2.94E–14	2.05E–05	6.03E–19	2.59E–16	9.90E–13	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	4.03E–08	^{223}Ra	1.20E–09	2.05E–05	2.46E–14	1.05E–11	4.03E–08	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	2.42E–06	^{224}Ra	7.18E–08	2.05E–05	1.47E–12	6.32E–10	2.42E–06	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ra	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	9.68E–09	^{226}Ra	2.88E–10	2.05E–05	5.90E–15	2.53E–12	9.68E–09	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ra	1.73E–14	2.05E–05	3.54E–19	1.52E–16	5.82E–13	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	2.41E–08	^{87}Rb	7.15E–10	2.05E–05	1.46E–14	6.28E–12	2.41E–08	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	8.29E–08	^{102}Rh	2.46E–09	2.05E–05	5.05E–14	2.17E–11	8.30E–08	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Rh	7.69E–09	2.05E–05	1.58E–13	6.76E–11	2.59E–07	0.00%

Table A-3. (continued).

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{219}Rn	1.20E–09	1.8	2.16E–09	4.03E–08	^{219}Rn	1.20E–09	2.05E–05	2.46E–14	1.05E–11	4.03E–08	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	2.42E–06	^{220}Rn	7.18E–08	2.05E–05	1.47E–12	6.32E–10	2.42E–06	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	9.68E–09	^{222}Rn	2.88E–10	2.05E–05	5.90E–15	2.53E–12	9.68E–09	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Ru	7.69E–09	2.05E–05	1.58E–13	6.76E–11	2.59E–07	0.00%
^{125}Sb	[d]		5.55E–04	1.04E–02	^{125}Sb	5.87E–05	2.05E–05	1.20E–09	5.16E–07	1.04E–02	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	4.76E–05	^{126}Sb	1.41E–06	2.05E–05	2.90E–11	1.24E–08	4.76E–05	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	3.40E–04	^{126m}Sb	1.01E–05	2.05E–05	2.07E–10	8.89E–08	3.40E–04	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	3.61E–04	^{79}Se	1.07E–05	2.05E–05	2.20E–10	9.44E–08	3.61E–04	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	1.56E–10	^{146}Sm	4.65E–12	2.05E–05	9.52E–17	4.09E–14	1.56E–10	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	6.10E–09	^{147}Sm	1.81E–10	2.05E–05	3.72E–15	1.59E–12	6.10E–09	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	3.13E–14	^{148}Sm	9.30E–16	2.05E–05	1.91E–20	8.18E–18	3.13E–14	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	2.78E–15	^{149}Sm	8.26E–17	2.05E–05	1.69E–21	7.26E–19	2.78E–15	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	2.59E–01	^{151}Sm	7.71E–03	2.05E–05	1.58E–07	6.78E–05	2.59E–01	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	3.21E–13	^{119m}Sn	9.55E–15	2.05E–05	1.96E–19	8.40E–17	3.21E–13	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	1.93E–03	^{121m}Sn	5.74E–05	2.05E–05	1.18E–09	5.05E–07	1.93E–03	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	3.40E–04	^{126}Sn	1.01E–05	2.05E–05	2.07E–10	8.89E–08	3.40E–04	0.00%
^{90}Sr	[d]		1.87E–02	3.50E–01	^{90}Sr	[e]		1.49E–05	6.39E–03	3.56E–01	0.99%
^{98}Tc	6.34E–11	1.8	1.14E–10	2.14E–09	^{98}Tc	6.34E–11	2.05E–05	1.30E–15	5.58E–13	2.14E–09	0.00%
^{99}Tc	[d]		6.17E–04	1.15E–02	^{99}Tc	[e]		5.25E–09	2.25E–06	1.15E–02	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	1.72E–15	^{123}Te	5.12E–17	2.05E–05	1.05E–21	4.51E–19	1.72E–15	0.00%
^{125m}Te	1.43E–05		1.36E–04	2.54E–03	^{125m}Te	1.43E–05	2.05E–05	2.94E–10	1.26E–07	2.54E–03	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	3.98E–08	^{227}Th	1.18E–09	2.05E–05	2.42E–14	1.04E–11	3.98E–08	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	2.41E–06	^{228}Th	7.16E–08	2.05E–05	1.47E–12	6.30E–10	2.41E–06	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	1.89E–10	^{229}Th	5.61E–12	2.05E–05	1.15E–16	4.93E–14	1.89E–10	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	7.92E–07	^{230}Th	2.35E–08	2.05E–05	4.82E–13	2.07E–10	7.92E–07	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	2.67E–05	^{231}Th	7.93E–07	2.05E–05	1.63E–11	6.97E–09	2.67E–05	0.00%

Table A-3. (continued).

Solids					Liquid					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{232}Th	1.85E–14	1.8	3.33E–14	6.23E–13	^{232}Th	1.85E–14	2.05E–05	3.79E–19	1.63E–16	6.23E–13	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	2.75E–05	^{234}Th	8.17E–07	2.05E–05	1.67E–11	7.18E–09	2.75E–05	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	4.02E–08	^{207}Tl	1.19E–09	2.05E–05	2.45E–14	1.05E–11	4.02E–08	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	8.69E–07	^{208}Tl	2.58E–08	2.05E–05	5.29E–13	2.27E–10	8.69E–07	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	4.08E–12	^{209}Tl	1.21E–13	2.05E–05	2.48E–18	1.07E–15	4.08E–12	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	1.61E–14	^{171}Tm	4.80E–16	2.05E–05	9.83E–21	4.22E–18	1.61E–14	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	2.33E–06	^{232}U	6.91E–08	2.05E–05	1.42E–12	6.08E–10	2.33E–06	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	5.97E–08	^{233}U	1.77E–09	2.05E–05	3.64E–14	1.56E–11	5.97E–08	0.00%
^{234}U	[d]		2.98E–06	5.57E–05	^{234}U	[e]		7.11E–11	3.05E–08	5.58E–05	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	2.67E–05	^{235}U	[e]		9.60E–12	4.12E–09	2.67E–05	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	6.22E–05	^{236}U	1.85E–06	2.05E–05	3.79E–11	1.63E–08	6.23E–05	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	7.23E–06	^{237}U	2.15E–07	2.05E–05	4.40E–12	1.89E–09	7.23E–06	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	2.75E–05	^{238}U	8.17E–07	2.05E–05	1.67E–11	7.18E–09	2.75E–05	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	9.88E–13	^{240}U	2.94E–14	2.05E–05	6.02E–19	2.58E–16	9.89E–13	0.00%
^{90}Y	[d]		1.87E–02	3.50E–01	^{90}Y	8.88E–01	2.05E–05	1.82E–05	7.81E–03	3.58E–01	0.99%
^{93}Zr	1.06E–04	1.8	1.90E–04	3.56E–03	^{93}Zr	1.06E–04	2.05E–05	2.17E–09	9.30E–07	3.56E–03	0.01%
			Total	36.12				Total	0.03	36.15	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-105.

Table A-4. Post-decontamination estimated inventory for Tank WM-106.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ac	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	4.02E–08	^{227}Ac	1.20E–09	6.91E–05	8.26E–14	3.54E–11	4.03E–08	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ac	1.73E–14	6.91E–05	1.19E–18	5.12E–16	5.82E–13	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	9.59E–12	^{108}Ag	2.85E–13	6.91E–05	1.97E–17	8.45E–15	9.60E–12	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.08E–10	^{108m}Ag	3.20E–12	6.91E–05	2.21E–16	9.49E–14	1.08E–10	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	8.00E–16	^{109m}Ag	2.38E–17	6.91E–05	1.64E–21	7.05E–19	8.01E–16	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	3.01E–17	^{110}Ag	8.93E–19	6.91E–05	6.17E–23	2.65E–20	3.01E–17	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	2.26E–15	^{110m}Ag	6.71E–17	6.91E–05	4.64E–21	1.99E–18	2.26E–15	0.00%
^{241}Am	[d]		3.40E–04	6.36E–03	^{241}Am	[e]		4.31E–09	1.85E–06	6.36E–03	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	2.40E–05	^{242}Am	7.13E–07	6.91E–05	4.93E–11	2.12E–08	2.40E–05	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	2.41E–05	^{242m}Am	7.17E–07	6.91E–05	4.95E–11	2.13E–08	2.42E–05	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	3.31E–05	^{243}Am	9.83E–07	6.91E–05	6.79E–11	2.91E–08	3.31E–05	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	1.89E–10	^{217}At	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{137m}Ba	[d]		9.20E–01	1.72E+01	^{137m}Ba	[e]		6.91E–05	2.96E–02	1.72E+01	46.90%
^{10}Be	7.56E–11	1.8	1.36E–10	2.55E–09	^{10}Be	7.56E–11	6.91E–05	5.23E–15	2.24E–12	2.55E–09	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Bi	1.11E–10	6.91E–05	7.70E–15	3.30E–12	3.75E–09	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	1.82E–22	^{210m}Bi	5.41E–24	6.91E–05	3.74E–28	1.61E–25	1.82E–22	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Bi	1.20E–09	6.91E–05	8.28E–14	3.55E–11	4.03E–08	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Bi	7.18E–08	6.91E–05	4.96E–12	2.13E–09	2.42E–06	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	1.89E–10	^{213}Bi	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Bi	2.88E–10	6.91E–05	1.99E–14	8.52E–12	9.69E–09	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	7.41E–08	^{14}C	[e]		1.23E–12	5.28E–10	7.46E–08	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	8.00E–16	^{109}Cd	2.38E–17	6.91E–05	1.64E–21	7.05E–19	8.01E–16	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	1.95E–03	^{113m}Cd	5.78E–05	6.91E–05	3.99E–09	1.71E–06	1.95E–03	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	2.46E–08	^{142}Ce	7.31E–10	6.91E–05	5.05E–14	2.17E–11	2.46E–08	0.00%

Table A-4. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{144}Ce	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Ce	2.33E–10	6.91E–05	1.61E–14	6.90E–12	7.84E–09	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	2.43E–14	^{249}Cf	7.21E–16	6.91E–05	4.98E–20	2.14E–17	2.43E–14	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	1.26E–14	^{250}Cf	3.73E–16	6.91E–05	2.58E–20	1.11E–17	1.26E–14	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	3.85E–16	^{251}Cf	1.14E–17	6.91E–05	7.90E–22	3.39E–19	3.85E–16	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	1.63E–17	^{252}Cf	4.84E–19	6.91E–05	3.35E–23	1.44E–20	1.63E–17	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	1.99E–05	^{242}Cm	5.91E–07	6.91E–05	4.09E–11	1.75E–08	1.99E–05	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	4.33E–06	^{243}Cm	1.29E–07	6.91E–05	8.88E–12	3.81E–09	4.33E–06	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	2.37E–04	^{244}Cm	[e]		1.60E–11	6.86E–09	2.37E–04	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	5.66E–08	^{245}Cm	1.68E–09	6.91E–05	1.16E–13	4.98E–11	5.66E–08	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	3.71E–09	^{246}Cm	1.10E–10	6.91E–05	7.61E–15	3.27E–12	3.71E–09	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	4.11E–15	^{247}Cm	1.22E–16	6.91E–05	8.43E–21	3.62E–18	4.11E–15	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	4.34E–15	^{248}Cm	1.29E–16	6.91E–05	8.90E–21	3.82E–18	4.34E–15	0.00%
^{60}Co	[d]		5.02E–05	9.39E–04	^{60}Co	[e]		6.91E–09	2.96E–06	9.42E–04	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.02E–03	^{134}Cs	3.03E–05	6.91E–05	2.10E–09	8.99E–07	1.02E–03	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	4.84E–04	^{135}Cs	1.44E–05	6.91E–05	9.94E–10	4.26E–07	4.85E–04	0.00%
^{137}Cs	[d]		9.20E–01	1.72E+01	^{137}Cs	[e]		6.91E–05	2.96E–02	1.72E+01	46.90%
^{150}Eu	2.96E–10	1.8	5.34E–10	9.98E–09	^{150}Eu	2.96E–10	6.91E–05	2.05E–14	8.79E–12	9.99E–09	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	1.32E–03	^{152}Eu	3.92E–05	6.91E–05	2.71E–09	1.16E–06	1.32E–03	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	5.88E–02	^{154}Eu	[e]		1.31E–08	5.62E–06	5.88E–02	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	1.59E–02	^{155}Eu	4.74E–04	6.91E–05	3.27E–08	1.40E–05	1.60E–02	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	1.64E–02	^{55}Fe	4.88E–04	6.91E–05	3.37E–08	1.45E–05	1.64E–02	0.04%
^{221}Fr	5.61E–12	1.8	1.01E–11	1.89E–10	^{221}Fr	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	5.55E–10	^{223}Fr	1.65E–11	6.91E–05	1.14E–15	4.89E–13	5.56E–10	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	1.20E–15	^{152}Gd	3.58E–17	6.91E–05	2.47E–21	1.06E–18	1.21E–15	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	1.40E–17	^{153}Gd	4.15E–19	6.91E–05	2.87E–23	1.23E–20	1.40E–17	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.08E–02	^{3}H	[e]		9.10E–09	3.90E–06	1.08E–02	0.03%

Table A-4. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{166m}Ho	1.13E–09	1.8	2.03E–09	3.79E–08	^{166m}Ho	1.13E–09	6.91E–05	7.78E–14	3.34E–11	3.79E–08	0.00%
^{129}I	[d]		6.24E–07	1.17E–05	^{129}I	[e]		6.95E–11	2.98E–08	1.17E–05	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	9.26E–15	^{115}In	2.75E–16	6.91E–05	1.90E–20	8.15E–18	9.27E–15	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	1.58E–13	^{138}La	4.68E–15	6.91E–05	3.24E–19	1.39E–16	1.58E–13	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	2.94E–03	^{93m}Nb	8.74E–05	6.91E–05	6.04E–09	2.59E–06	2.94E–03	0.01%
^{94}Nb	[d]		1.66E–04	3.10E–03	^{94}Nb	[e]		4.37E–08	1.87E–05	3.12E–03	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	1.33E–12	^{144}Nd	3.96E–14	6.91E–05	2.74E–18	1.18E–15	1.34E–12	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	3.79E–04	^{59}Ni	1.13E–05	6.91E–05	7.79E–10	3.34E–07	3.80E–04	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	4.32E–02	^{63}Ni	[e]		2.04E–08	8.75E–06	4.32E–02	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	7.42E–09	^{236}Np	2.21E–10	6.91E–05	1.52E–14	6.54E–12	7.43E–09	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	7.10E–04	^{237}Np	[e]		8.18E–11	3.51E–08	7.10E–04	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	1.21E–07	^{238}Np	3.58E–09	6.91E–05	2.48E–13	1.06E–10	1.21E–07	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	3.31E–05	^{239}Np	9.83E–07	6.91E–05	6.79E–11	2.91E–08	3.31E–05	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	9.88E–13	^{240m}Np	2.94E–14	6.91E–05	2.03E–18	8.70E–16	9.89E–13	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	7.14E–08	^{231}Pa	2.12E–09	6.91E–05	1.47E–13	6.29E–11	7.14E–08	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	7.10E–04	^{233}Pa	2.11E–05	6.91E–05	1.46E–09	6.26E–07	7.11E–04	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	3.57E–08	^{234}Pa	1.06E–09	6.91E–05	7.33E–14	3.15E–11	3.58E–08	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	2.75E–05	^{234m}Pa	8.17E–07	6.91E–05	5.64E–11	2.42E–08	2.75E–05	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	1.89E–10	^{209}Pb	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	3.75E–09	^{210}Pb	1.11E–10	6.91E–05	7.70E–15	3.30E–12	3.75E–09	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	4.03E–08	^{211}Pb	1.20E–09	6.91E–05	8.28E–14	3.55E–11	4.03E–08	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	2.42E–06	^{212}Pb	7.18E–08	6.91E–05	4.96E–12	2.13E–09	2.42E–06	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	9.68E–09	^{214}Pb	2.88E–10	6.91E–05	1.99E–14	8.52E–12	9.69E–09	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	1.37E–05	^{107}Pd	4.06E–07	6.91E–05	2.80E–11	1.20E–08	1.37E–05	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	1.35E–05	^{146}Pm	4.00E–07	6.91E–05	2.76E–11	1.19E–08	1.35E–05	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	1.31E–02	^{147}Pm	3.88E–04	6.91E–05	2.68E–08	1.15E–05	1.31E–02	0.04%

Table A-4. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{210}Po	1.08E–10	1.8	1.94E–10	3.62E–09	^{210}Po	1.08E–10	6.91E–05	7.44E–15	3.19E–12	3.63E–09	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	1.13E–10	^{211}Po	3.35E–12	6.91E–05	2.32E–16	9.94E–14	1.13E–10	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	1.55E–06	^{212}Po	4.60E–08	6.91E–05	3.18E–12	1.36E–09	1.55E–06	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	1.85E–10	^{213}Po	5.49E–12	6.91E–05	3.79E–16	1.63E–13	1.85E–10	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	9.68E–09	^{214}Po	2.87E–10	6.91E–05	1.99E–14	8.52E–12	9.68E–09	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	4.03E–08	^{215}Po	1.20E–09	6.91E–05	8.28E–14	3.55E–11	4.03E–08	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	2.42E–06	^{216}Po	7.18E–08	6.91E–05	4.96E–12	2.13E–09	2.42E–06	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	9.68E–09	^{218}Po	2.88E–10	6.91E–05	1.99E–14	8.52E–12	9.69E–09	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	7.83E–09	^{144}Pr	2.33E–10	6.91E–05	1.61E–14	6.90E–12	7.84E–09	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	9.40E–11	^{144m}Pr	2.79E–12	6.91E–05	1.93E–16	8.28E–14	9.41E–11	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	2.20E–07	^{236}Pu	6.52E–09	6.91E–05	4.51E–13	1.93E–10	2.20E–07	0.00%
^{238}Pu	[d]		9.23E–03	1.73E–01	^{238}Pu	[e]		3.84E–08	1.65E–05	1.73E–01	0.47%
^{239}Pu	[d]		2.75E–03	5.14E–02	^{239}Pu	[e]		6.19E–09	2.66E–06	5.14E–02	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	2.04E–02	^{240}Pu	6.06E–04	6.91E–05	4.19E–08	1.80E–05	2.04E–02	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	2.95E–01	^{241}Pu	[e]		2.84E–08	1.22E–05	2.95E–01	0.80%
^{242}Pu	4.43E–07	1.8	7.98E–07	1.49E–05	^{242}Pu	4.43E–07	6.91E–05	3.06E–11	1.31E–08	1.49E–05	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	4.11E–15	^{243}Pu	1.22E–16	6.91E–05	8.43E–21	3.62E–18	4.11E–15	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	9.90E–13	^{244}Pu	2.94E–14	6.91E–05	2.03E–18	8.72E–16	9.90E–13	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	4.03E–08	^{223}Ra	1.20E–09	6.91E–05	8.28E–14	3.55E–11	4.03E–08	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	2.42E–06	^{224}Ra	7.18E–08	6.91E–05	4.96E–12	2.13E–09	2.42E–06	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	1.89E–10	^{225}Ra	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	9.68E–09	^{226}Ra	2.88E–10	6.91E–05	1.99E–14	8.52E–12	9.69E–09	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	5.81E–13	^{228}Ra	1.73E–14	6.91E–05	1.19E–18	5.12E–16	5.82E–13	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	2.41E–08	^{87}Rb	7.15E–10	6.91E–05	4.94E–14	2.12E–11	2.41E–08	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	8.29E–08	^{102}Rh	2.46E–09	6.91E–05	1.70E–13	7.30E–11	8.30E–08	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Rh	7.69E–09	6.91E–05	5.31E–13	2.28E–10	2.59E–07	0.00%

Table A-4. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{219}Rn	1.20E–09	1.8	2.16E–09	4.03E–08	^{219}Rn	1.20E–09	6.91E–05	8.28E–14	3.55E–11	4.03E–08	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	2.42E–06	^{220}Rn	7.18E–08	6.91E–05	4.96E–12	2.13E–09	2.42E–06	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	9.68E–09	^{222}Rn	2.88E–10	6.91E–05	1.99E–14	8.52E–12	9.69E–09	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	2.59E–07	^{106}Ru	7.69E–09	6.91E–05	5.31E–13	2.28E–10	2.59E–07	0.00%
^{125}Sb	[d]		5.55E–04	1.04E–02	^{125}Sb	5.87E–05	6.91E–05	4.06E–09	1.74E–06	1.04E–02	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	4.76E–05	^{126}Sb	1.41E–06	6.91E–05	9.78E–11	4.19E–08	4.77E–05	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	3.40E–04	^{126m}Sb	1.01E–05	6.91E–05	6.98E–10	3.00E–07	3.40E–04	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	3.61E–04	^{79}Se	1.07E–05	6.91E–05	7.41E–10	3.18E–07	3.62E–04	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	1.56E–10	^{146}Sm	4.65E–12	6.91E–05	3.21E–16	1.38E–13	1.57E–10	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	6.10E–09	^{147}Sm	1.81E–10	6.91E–05	1.25E–14	5.37E–12	6.11E–09	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	3.13E–14	^{148}Sm	9.30E–16	6.91E–05	6.43E–20	2.76E–17	3.13E–14	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	2.78E–15	^{149}Sm	8.26E–17	6.91E–05	5.71E–21	2.45E–18	2.78E–15	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	2.59E–01	^{151}Sm	7.71E–03	6.91E–05	5.33E–07	2.28E–04	2.60E–01	0.71%
^{119m}Sn	9.55E–15	1.8	1.72E–14	3.21E–13	^{119m}Sn	9.55E–15	6.91E–05	6.60E–19	2.83E–16	3.22E–13	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	1.93E–03	^{121m}Sn	5.74E–05	6.91E–05	3.96E–09	1.70E–06	1.93E–03	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	3.40E–04	^{126}Sn	1.01E–05	6.91E–05	6.98E–10	3.00E–07	3.40E–04	0.00%
^{90}Sr	[d]		1.87E–02	3.50E–01	^{90}Sr	[e]		6.61E–04	2.84E–01	6.33E–01	1.72%
^{98}Tc	6.34E–11	1.8	1.14E–10	2.14E–09	^{98}Tc	6.34E–11	6.91E–05	4.38E–15	1.88E–12	2.14E–09	0.00%
^{99}Tc	[d]		6.17E–04	1.15E–02	^{99}Tc	[e]		1.94E–07	8.32E–05	1.16E–02	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	1.72E–15	^{123}Te	5.12E–17	6.91E–05	3.54E–21	1.52E–18	1.73E–15	0.00%
^{125m}Te	1.43E–05		1.36E–04	2.54E–03	^{125m}Te	1.43E–05	6.91E–05	9.90E–10	4.25E–07	2.54E–03	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	3.98E–08	^{227}Th	1.18E–09	6.91E–05	8.16E–14	3.50E–11	3.98E–08	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	2.41E–06	^{228}Th	7.16E–08	6.91E–05	4.95E–12	2.12E–09	2.41E–06	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	1.89E–10	^{229}Th	5.61E–12	6.91E–05	3.87E–16	1.66E–13	1.89E–10	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	7.92E–07	^{230}Th	2.35E–08	6.91E–05	1.63E–12	6.98E–10	7.93E–07	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	2.67E–05	^{231}Th	7.93E–07	6.91E–05	5.48E–11	2.35E–08	2.67E–05	0.00%

Table A-4. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{232}Th	1.85E–14	1.8	3.33E–14	6.23E–13	^{232}Th	1.85E–14	6.91E–05	1.28E–18	5.48E–16	6.23E–13	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	2.75E–05	^{234}Th	8.17E–07	6.91E–05	5.64E–11	2.42E–08	2.75E–05	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	4.02E–08	^{207}Tl	1.19E–09	6.91E–05	8.25E–14	3.54E–11	4.02E–08	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	8.69E–07	^{208}Tl	2.58E–08	6.91E–05	1.78E–12	7.65E–10	8.70E–07	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	4.08E–12	^{209}Tl	1.21E–13	6.91E–05	8.37E–18	3.59E–15	4.08E–12	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	1.61E–14	^{171}Tm	4.80E–16	6.91E–05	3.31E–20	1.42E–17	1.62E–14	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	2.33E–06	^{232}U	6.91E–08	6.91E–05	4.77E–12	2.05E–09	2.33E–06	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	5.97E–08	^{233}U	1.77E–09	6.91E–05	1.23E–13	5.26E–11	5.98E–08	0.00%
^{234}U	[d]		2.98E–06	5.57E–05	^{234}U	[e]		7.08E–11	3.04E–08	5.58E–05	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	2.67E–05	^{235}U	[e]		7.74E–12	3.32E–09	2.67E–05	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	6.22E–05	^{236}U	1.85E–06	6.91E–05	1.28E–10	5.48E–08	6.23E–05	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	7.23E–06	^{237}U	2.15E–07	6.91E–05	1.48E–11	6.36E–09	7.23E–06	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	2.75E–05	^{238}U	8.17E–07	6.91E–05	5.64E–11	2.42E–08	2.75E–05	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	9.88E–13	^{240}U	2.94E–14	6.91E–05	2.03E–18	8.70E–16	9.89E–13	0.00%
^{90}Y	[d]		1.87E–02	3.50E–01	^{90}Y	8.88E–01	6.91E–05	6.14E–05	2.63E–02	3.76E–01	1.72%
^{93}Zr	1.06E–04	1.8	1.90E–04	3.56E–03	^{93}Zr	1.06E–04	6.91E–05	7.31E–09	3.13E–06	3.56E–03	0.01%
			Total	36.1				Total	0.6	36.7	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-106.

Table A-5. Post-decontamination estimated inventory for Tank WM-180.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	5.47E–09	^{225}Ac	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	1.17E–06	^{227}Ac	1.20E–09	4.38E–07	5.24E–16	2.61E–12	1.17E–06	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	1.68E–11	^{228}Ac	1.73E–14	4.38E–07	7.57E–21	3.77E–17	1.68E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	2.78E–10	^{108}Ag	2.85E–13	4.38E–07	1.25E–19	6.23E–16	2.78E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	3.12E–09	^{108m}Ag	3.20E–12	4.38E–07	1.40E–18	7.00E–15	3.12E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	2.32E–14	^{109m}Ag	2.38E–17	4.38E–07	1.04E–23	5.20E–20	2.32E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	8.71E–16	^{110}Ag	8.93E–19	4.38E–07	3.91E–25	1.95E–21	8.71E–16	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	6.55E–14	^{110m}Ag	6.71E–17	4.38E–07	2.94E–23	1.47E–19	6.55E–14	0.00%
^{241}Am	[d]		3.40E–04	1.84E–01	^{241}Am	[e]		6.63E–10	3.31E–06	1.84E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	6.96E–04	^{242}Am	7.13E–07	4.38E–07	3.13E–13	1.56E–09	6.96E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	6.99E–04	^{242m}Am	7.17E–07	4.38E–07	3.14E–13	1.57E–09	6.99E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	9.59E–04	^{243}Am	9.83E–07	4.38E–07	4.31E–13	2.15E–09	9.59E–04	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	5.47E–09	^{217}At	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{137m}Ba	[d]		9.20E–01	4.99E+02	^{137m}Ba	[e]		4.38E–07	2.19E–03	4.99E+02	47.64%
^{10}Be	7.56E–11	1.8	1.36E–10	7.38E–08	^{10}Be	7.56E–11	4.38E–07	3.31E–17	1.65E–13	7.38E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	1.09E–07	^{210}Bi	1.11E–10	4.38E–07	4.88E–17	2.44E–13	1.09E–07	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	5.28E–21	^{210m}Bi	5.41E–24	4.38E–07	2.37E–30	1.18E–26	5.28E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	1.17E–06	^{211}Bi	1.20E–09	4.38E–07	5.25E–16	2.62E–12	1.17E–06	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	7.01E–05	^{212}Bi	7.18E–08	4.38E–07	3.15E–14	1.57E–10	7.01E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	5.47E–09	^{213}Bi	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	2.80E–07	^{214}Bi	2.88E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	2.15E–06	^{14}C	2.2E–09	4.38E–07	9.64E–16	4.81E–12	2.15E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	2.32E–14	^{109}Cd	2.38E–17	4.38E–07	1.04E–23	5.20E–20	2.32E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	5.64E–02	^{113m}Cd	5.78E–05	4.38E–07	2.53E–11	1.26E–07	5.64E–02	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	7.13E–07	^{142}Ce	7.31E–10	4.38E–07	3.20E–16	1.60E–12	7.13E–07	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	2.27E–07	^{144}Ce	2.33E–10	4.38E–07	1.02E–16	5.08E–13	2.27E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	7.03E–13	^{249}Cf	7.21E–16	4.38E–07	3.16E–22	1.57E–18	7.03E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	3.64E–13	^{250}Cf	3.73E–16	4.38E–07	1.63E–22	8.15E–19	3.64E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	1.11E–14	^{251}Cf	1.14E–17	4.38E–07	5.00E–24	2.50E–20	1.11E–14	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	4.72E–16	^{252}Cf	4.84E–19	4.38E–07	2.12E–25	1.06E–21	4.72E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	5.77E–04	^{242}Cm	5.91E–07	4.38E–07	2.59E–13	1.29E–09	5.77E–04	0.00%

Table A-5. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{243}Cm	1.29E–07	1.8	2.31E–07	1.25E–04	^{243}Cm	1.29E–07	4.38E–07	5.63E–14	2.81E–10	1.25E–04	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	6.86E–03	^{244}Cm	[e]		2.84E–11	1.42E–07	6.86E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	1.64E–06	^{245}Cm	1.68E–09	4.38E–07	7.36E–16	3.67E–12	1.64E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	1.07E–07	^{246}Cm	1.10E–10	4.38E–07	4.82E–17	2.41E–13	1.07E–07	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	1.19E–13	^{247}Cm	1.22E–16	4.38E–07	5.34E–23	2.67E–19	1.19E–13	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	1.26E–13	^{248}Cm	1.29E–16	4.38E–07	5.64E–23	2.81E–19	1.26E–13	0.00%
^{60}Co	[d]		5.02E–05	2.72E–02	^{60}Co	[e]		2.14E–10	1.07E–06	2.72E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	2.96E–02	^{134}Cs	3.03E–05	4.38E–07	1.33E–11	6.63E–08	2.96E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	1.40E–02	^{135}Cs	1.44E–05	4.38E–07	6.30E–12	3.14E–08	1.40E–02	0.00%
^{137}Cs	[d]		9.20E–01	4.99E+02	^{137}Cs	[e]		4.38E–07	2.19E–03	4.99E+02	47.64%
^{150}Eu	2.96E–10	1.8	5.34E–10	2.89E–07	^{150}Eu	2.96E–10	4.38E–07	1.30E–16	6.48E–13	2.89E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	3.82E–02	^{152}Eu	3.92E–05	4.38E–07	1.72E–11	8.56E–08	3.82E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	1.70E+00	^{154}Eu	[e]		7.26E–11	3.62E–07	1.70E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	4.62E–01	^{155}Eu	4.74E–04	4.38E–07	2.07E–10	1.04E–06	4.62E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	4.76E–01	^{55}Fe	4.88E–04	4.38E–07	2.14E–10	1.07E–06	4.76E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	5.47E–09	^{221}Fr	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	1.61E–08	^{223}Fr	1.65E–11	4.38E–07	7.23E–18	3.60E–14	1.61E–08	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	3.49E–14	^{152}Gd	3.58E–17	4.38E–07	1.57E–23	7.82E–20	3.49E–14	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	4.05E–16	^{153}Gd	4.15E–19	4.38E–07	1.82E–25	9.08E–22	4.05E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	3.14E–01	^{3}H	3.22E–04	4.38E–07	1.41E–10	7.03E–07	3.14E–01	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	1.10E–06	^{166m}Ho	1.13E–09	4.38E–07	4.93E–16	2.46E–12	1.10E–06	0.00%
^{129}I	[d]		6.24E–07	3.38E–04	^{129}I	5.88E–07	4.38E–07	2.58E–13	1.29E–09	3.38E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	2.68E–13	^{115}In	2.75E–16	4.38E–07	1.20E–22	6.01E–19	2.68E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	4.57E–12	^{138}La	4.68E–15	4.38E–07	2.05E–21	1.02E–17	4.57E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	8.52E–02	^{93m}Nb	8.74E–05	4.38E–07	3.83E–11	1.91E–07	8.52E–02	0.01%
^{94}Nb	[d]		1.66E–04	9.00E–02	^{94}Nb	[e]		4.67E–11	2.33E–07	9.00E–02	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	3.87E–11	^{144}Nd	3.96E–14	4.38E–07	1.74E–20	8.66E–17	3.87E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	1.10E–02	^{59}Ni	1.13E–05	4.38E–07	4.94E–12	2.46E–08	1.10E–02	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	1.25E+00	^{63}Ni	[e]		1.07E–10	5.34E–07	1.25E+00	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	2.15E–07	^{236}Np	2.21E–10	4.38E–07	9.66E–17	4.82E–13	2.15E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	2.06E–02	^{237}Np	[e]		1.35E–10	6.74E–07	2.06E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	3.50E–06	^{238}Np	3.58E–09	4.38E–07	1.57E–15	7.83E–12	3.50E–06	0.00%

Table A-5. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{239}Np	9.83E–07	1.8	1.77E–06	9.59E–04	^{239}Np	9.83E–07	4.38E–07	4.31E–13	2.15E–09	9.59E–04	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	2.86E–11	^{240m}Np	2.94E–14	4.38E–07	1.29E–20	6.42E–17	2.86E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	2.07E–06	^{231}Pa	2.12E–09	4.38E–07	9.29E–16	4.63E–12	2.07E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	2.06E–02	^{233}Pa	2.11E–05	4.38E–07	9.24E–12	4.61E–08	2.06E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	1.04E–06	^{234}Pa	1.06E–09	4.38E–07	4.65E–16	2.32E–12	1.04E–06	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	7.97E–04	^{234m}Pa	8.17E–07	4.38E–07	3.58E–13	1.78E–09	7.97E–04	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	5.47E–09	^{209}Pb	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	1.09E–07	^{210}Pb	1.11E–10	4.38E–07	4.88E–17	2.43E–13	1.09E–07	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	1.17E–06	^{211}Pb	1.20E–09	4.38E–07	5.25E–16	2.62E–12	1.17E–06	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	7.01E–05	^{212}Pb	7.18E–08	4.38E–07	3.15E–14	1.57E–10	7.01E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	2.80E–07	^{214}Pb	2.88E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	3.96E–04	^{107}Pd	4.06E–07	4.38E–07	1.78E–13	8.87E–10	3.96E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	3.90E–04	^{146}Pm	4.00E–07	4.38E–07	1.75E–13	8.74E–10	3.90E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	3.79E–01	^{147}Pm	3.88E–04	4.38E–07	1.70E–10	8.49E–07	3.79E–01	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	1.05E–07	^{210}Po	1.08E–10	4.38E–07	4.71E–17	2.35E–13	1.05E–07	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	3.27E–09	^{211}Po	3.35E–12	4.38E–07	1.47E–18	7.33E–15	3.27E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	4.49E–05	^{212}Po	4.60E–08	4.38E–07	2.02E–14	1.01E–10	4.49E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	5.35E–09	^{213}Po	5.49E–12	4.38E–07	2.40E–18	1.20E–14	5.35E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	2.80E–07	^{214}Po	2.87E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	1.17E–06	^{215}Po	1.20E–09	4.38E–07	5.25E–16	2.62E–12	1.17E–06	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	7.01E–05	^{216}Po	7.18E–08	4.38E–07	3.15E–14	1.57E–10	7.01E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	2.80E–07	^{218}Po	2.88E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	2.27E–07	^{144}Pr	2.33E–10	4.38E–07	1.02E–16	5.08E–13	2.27E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	2.72E–09	^{144m}Pr	2.79E–12	4.38E–07	1.22E–18	6.10E–15	2.72E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	6.36E–06	^{236}Pu	6.52E–09	4.38E–07	2.86E–15	1.43E–11	6.36E–06	0.00%
^{238}Pu	[d]		9.23E–03	5.00E+00	^{238}Pu	[e]		3.77E–08	1.88E–04	5.00E+00	0.48%
^{239}Pu	[d]		2.75E–03	1.49E+00	^{239}Pu	[e]		6.99E–09	3.49E–05	1.49E+00	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	5.91E–01	^{240}Pu	6.06E–04	4.38E–07	2.65E–10	1.32E–06	5.91E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	8.53E+00	^{241}Pu	[e]		3.45E–08	1.72E–04	8.53E+00	0.82%
^{242}Pu	4.43E–07	1.8	7.98E–07	4.32E–04	^{242}Pu	4.43E–07	4.38E–07	1.94E–13	9.68E–10	4.32E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	1.19E–13	^{243}Pu	1.22E–16	4.38E–07	5.34E–23	2.67E–19	1.19E–13	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	2.87E–11	^{244}Pu	2.94E–14	4.38E–07	1.29E–20	6.42E–17	2.87E–11	0.00%

Table A-5. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{223}Ra	1.20E–09	1.8	2.16E–09	1.17E–06	^{223}Ra	1.20E–09	4.38E–07	5.25E–16	2.62E–12	1.17E–06	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	7.01E–05	^{224}Ra	7.18E–08	4.38E–07	3.15E–14	1.57E–10	7.01E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	5.47E–09	^{225}Ra	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	2.80E–07	^{226}Ra	2.88E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	1.68E–11	^{228}Ra	1.73E–14	4.38E–07	7.57E–21	3.77E–17	1.68E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	6.97E–07	^{87}Rb	7.15E–10	4.38E–07	3.13E–16	1.56E–12	6.97E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	2.40E–06	^{102}Rh	2.46E–09	4.38E–07	1.08E–15	5.38E–12	2.40E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	7.50E–06	^{106}Rh	7.69E–09	4.38E–07	3.37E–15	1.68E–11	7.50E–06	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	1.17E–06	^{219}Rn	1.20E–09	4.38E–07	5.25E–16	2.62E–12	1.17E–06	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	7.01E–05	^{220}Rn	7.18E–08	4.38E–07	3.15E–14	1.57E–10	7.01E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	2.80E–07	^{222}Rn	2.88E–10	4.38E–07	1.26E–16	6.28E–13	2.80E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	7.50E–06	^{106}Ru	7.69E–09	4.38E–07	3.37E–15	1.68E–11	7.50E–06	0.00%
^{125}Sb	[d]		5.55E–04	3.01E–01	^{125}Sb	[e]		8.62E–10	4.30E–06	3.01E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	1.38E–03	^{126}Sb	1.41E–06	4.38E–07	6.20E–13	3.09E–09	1.38E–03	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	9.86E–03	^{126m}Sb	1.01E–05	4.38E–07	4.43E–12	2.21E–08	9.86E–03	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	1.05E–02	^{79}Se	1.07E–05	4.38E–07	4.70E–12	2.34E–08	1.05E–02	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	4.53E–09	^{146}Sm	4.65E–12	4.38E–07	2.03E–18	1.02E–14	4.53E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	1.77E–07	^{147}Sm	1.81E–10	4.38E–07	7.94E–17	3.96E–13	1.77E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	9.07E–13	^{148}Sm	9.30E–16	4.38E–07	4.07E–22	2.03E–18	9.07E–13	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	8.06E–14	^{149}Sm	8.26E–17	4.38E–07	3.62E–23	1.80E–19	8.06E–14	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	7.52E+00	^{151}Sm	7.71E–03	4.38E–07	3.38E–09	1.68E–05	7.52E+00	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	9.31E–12	^{119m}Sn	9.55E–15	4.38E–07	4.18E–21	2.09E–17	9.31E–12	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	5.60E–02	^{121m}Sn	5.74E–05	4.38E–07	2.51E–11	1.25E–07	5.60E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	9.86E–03	^{126}Sn	1.01E–05	4.38E–07	4.43E–12	2.21E–08	9.86E–03	0.00%
^{90}Sr	[d]		1.87E–02	1.01E+01	^{90}Sr	[e]		3.34E–08	1.67E–04	1.01E+01	0.97%
^{98}Tc	6.34E–11	1.8	1.14E–10	6.19E–08	^{98}Tc	6.34E–11	4.38E–07	2.78E–17	1.39E–13	6.19E–08	0.00%
^{99}Tc	[d]		6.17E–04	3.34E–01	^{99}Tc	[e]		3.56E–10	1.78E–06	3.34E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	5.00E–14	^{123}Te	5.12E–17	4.38E–07	2.24E–23	1.12E–19	5.00E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	7.37E–02	^{125m}Te	1.43E–05	4.38E–07	6.27E–12	3.13E–08	7.37E–02	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	1.15E–06	^{227}Th	1.18E–09	4.38E–07	5.17E–16	2.58E–12	1.15E–06	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	6.99E–05	^{228}Th	7.16E–08	4.38E–07	3.14E–14	1.57E–10	6.99E–05	0.00%

Table A-5. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{229}Th	5.61E–12	1.8	1.01E–11	5.47E–09	^{229}Th	5.61E–12	4.38E–07	2.46E–18	1.23E–14	5.47E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	2.30E–05	^{230}Th	2.35E–08	4.38E–07	1.03E–14	5.14E–11	2.30E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	7.73E–04	^{231}Th	7.93E–07	4.38E–07	3.47E–13	1.73E–09	7.73E–04	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	1.80E–11	^{232}Th	1.85E–14	4.38E–07	8.10E–21	4.04E–17	1.80E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	7.97E–04	^{234}Th	8.17E–07	4.38E–07	3.58E–13	1.78E–09	7.97E–04	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	1.17E–06	^{207}Tl	1.19E–09	4.38E–07	5.23E–16	2.61E–12	1.17E–06	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	2.52E–05	^{208}Tl	2.58E–08	4.38E–07	1.13E–14	5.64E–11	2.52E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	1.18E–10	^{209}Tl	1.21E–13	4.38E–07	5.31E–20	2.65E–16	1.18E–10	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	4.68E–13	^{171}Tm	4.80E–16	4.38E–07	2.10E–22	1.05E–18	4.68E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	6.74E–05	^{232}U	6.91E–08	4.38E–07	3.03E–14	1.51E–10	6.74E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	1.73E–06	^{233}U	1.77E–09	4.38E–07	7.77E–16	3.88E–12	1.73E–06	0.00%
^{234}U	[d]		2.98E–06	1.61E–03	^{234}U	[e]		6.41E–11	3.20E–07	1.62E–03	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	7.73E–04	^{235}U	7.93E–07	4.38E–07	3.47E–13	1.73E–09	7.73E–04	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	1.80E–03	^{236}U	1.85E–06	4.38E–07	8.10E–13	4.04E–09	1.80E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	2.09E–04	^{237}U	2.15E–07	4.38E–07	9.40E–14	4.69E–10	2.09E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	7.97E–04	^{238}U	8.17E–07	4.38E–07	3.58E–13	1.78E–09	7.97E–04	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	2.86E–11	^{240}U	2.94E–14	4.38E–07	1.29E–20	6.42E–17	2.86E–11	0.00%
^{90}Y	[d]		1.87E–02	1.01E+01	^{90}Y	8.88E–01	4.38E–07	3.89E–07	1.94E–03	1.01E+01	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	1.03E–01	^{93}Zr	1.06E–04	4.38E–07	4.63E–11	2.31E–07	1.03E–01	0.01%
			Total	1,046.56				Total	0.01	1,046.56	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-180.

Table A-6. Post-decontamination estimated inventory for Tank WM-181.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	2.48E–09	^{225}Ac	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	5.29E–07	^{227}Ac	1.20E–09	1.10E–05	1.31E–14	6.56E–11	5.29E–07	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	7.65E–12	^{228}Ac	1.73E–14	1.10E–05	1.90E–19	9.48E–16	7.65E–12	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	1.26E–10	^{108}Ag	2.85E–13	1.10E–05	3.13E–18	1.56E–14	1.26E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.42E–09	^{108m}Ag	3.20E–12	1.10E–05	3.52E–17	1.76E–13	1.42E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	1.05E–14	^{109m}Ag	2.38E–17	1.10E–05	2.62E–22	1.31E–18	1.05E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	3.95E–16	^{110}Ag	8.93E–19	1.10E–05	9.82E–24	4.90E–20	3.96E–16	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	2.97E–14	^{110m}Ag	6.71E–17	1.10E–05	7.38E–22	3.68E–18	2.97E–14	0.00%
^{241}Am	[d]		3.40E–04	8.37E–02	^{241}Am	[e]		1.19E–09	5.94E–06	8.37E–02	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	3.16E–04	^{242}Am	7.13E–07	1.10E–05	7.85E–12	3.92E–08	3.16E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	3.18E–04	^{242m}Am	7.17E–07	1.10E–05	7.89E–12	3.93E–08	3.18E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	4.35E–04	^{243}Am	9.83E–07	1.10E–05	1.08E–11	5.40E–08	4.36E–04	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	2.48E–09	^{217}At	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{137m}Ba	[d]		9.20E–01	2.26E+02	^{137m}Ba	[e]		1.10E–05	5.49E–02	2.26E+02	47.64%
^{10}Be	7.56E–11	1.8	1.36E–10	3.35E–08	^{10}Be	7.56E–11	1.10E–05	8.32E–16	4.15E–12	3.35E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	4.94E–08	^{210}Bi	1.11E–10	1.10E–05	1.23E–15	6.12E–12	4.94E–08	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	2.40E–21	^{210m}Bi	5.41E–24	1.10E–05	5.96E–29	2.97E–25	2.40E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	5.30E–07	^{211}Bi	1.20E–09	1.10E–05	1.32E–14	6.57E–11	5.31E–07	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	3.18E–05	^{212}Bi	7.18E–08	1.10E–05	7.90E–13	3.94E–09	3.18E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	2.48E–09	^{213}Bi	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	1.27E–07	^{214}Bi	2.88E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	9.74E–07	^{14}C	[e]		1.64E–11	8.18E–08	1.06E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	1.05E–14	^{109}Cd	2.38E–17	1.10E–05	2.62E–22	1.31E–18	1.05E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	2.56E–02	^{113m}Cd	5.78E–05	1.10E–05	6.36E–10	3.17E–06	2.56E–02	0.01%

Table A-6. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{142}Ce	7.31E–10	1.8	1.32E–09	3.24E–07	^{142}Ce	7.31E–10	1.10E–05	8.05E–15	4.01E–11	3.24E–07	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	1.03E–07	^{144}Ce	2.33E–10	1.10E–05	2.56E–15	1.28E–11	1.03E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	3.19E–13	^{249}Cf	7.21E–16	1.10E–05	7.93E–21	3.95E–17	3.19E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	1.65E–13	^{250}Cf	3.73E–16	1.10E–05	4.10E–21	2.05E–17	1.65E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	5.06E–15	^{251}Cf	1.14E–17	1.10E–05	1.26E–22	6.27E–19	5.06E–15	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	2.14E–16	^{252}Cf	4.84E–19	1.10E–05	5.33E–24	2.66E–20	2.14E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	2.62E–04	^{242}Cm	5.91E–07	1.10E–05	6.50E–12	3.24E–08	2.62E–04	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	5.69E–05	^{243}Cm	1.29E–07	1.10E–05	1.41E–12	7.05E–09	5.69E–05	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	3.12E–03	^{244}Cm	7.03E–06	1.10E–05	7.74E–11	3.86E–07	3.12E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	7.44E–07	^{245}Cm	1.68E–09	1.10E–05	1.85E–14	9.22E–11	7.44E–07	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	4.88E–08	^{246}Cm	1.10E–10	1.10E–05	1.21E–15	6.04E–12	4.88E–08	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	5.40E–14	^{247}Cm	1.22E–16	1.10E–05	1.34E–21	6.69E–18	5.40E–14	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	5.70E–14	^{248}Cm	1.29E–16	1.10E–05	1.42E–21	7.07E–18	5.71E–14	0.00%
^{60}Co	[d]		5.02E–05	1.24E–02	^{60}Co	[e]		2.94E–09	1.47E–05	1.24E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.34E–02	^{134}Cs	3.03E–05	1.10E–05	3.34E–10	1.66E–06	1.34E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	6.37E–03	^{135}Cs	1.44E–05	1.10E–05	1.58E–10	7.89E–07	6.37E–03	0.00%
^{137}Cs	[d]		9.20E–01	2.26E+02	^{137}Cs	[e]		1.10E–05	5.49E–02	2.26E+02	47.64%
^{150}Eu	2.96E–10	1.8	5.34E–10	1.31E–07	^{150}Eu	2.96E–10	1.10E–05	3.26E–15	1.63E–11	1.31E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	1.74E–02	^{152}Eu	3.92E–05	1.10E–05	4.31E–10	2.15E–06	1.74E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	7.74E–01	^{154}Eu	[e]		3.83E–09	1.91E–05	7.74E–01	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	2.10E–01	^{155}Eu	4.74E–04	1.10E–05	5.21E–09	2.60E–05	2.10E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	2.16E–01	^{55}Fe	4.88E–04	1.10E–05	5.37E–09	2.68E–05	2.16E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	2.48E–09	^{221}Fr	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	7.31E–09	^{223}Fr	1.65E–11	1.10E–05	1.81E–16	9.05E–13	7.31E–09	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	1.58E–14	^{152}Gd	3.58E–17	1.10E–05	3.94E–22	1.96E–18	1.58E–14	0.00%

Table A-6. (continued).

Solids					Liquids						Total (Solids+Liquids)
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{153}Gd	4.15E–19	1.8	7.48E–19	1.84E–16	^{153}Gd	4.15E–19	1.10E–05	4.57E–24	2.28E–20	1.84E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.42E–01	^{3}H	[e]		1.41E–09	7.03E–06	1.42E–01	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	4.99E–07	^{166m}Ho	1.13E–09	1.10E–05	1.24E–14	6.18E–11	4.99E–07	0.00%
^{129}I	[d]		6.24E–07	1.54E–04	^{129}I	[e]		7.41E–11	3.70E–07	1.54E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	1.22E–13	^{115}In	2.75E–16	1.10E–05	3.03E–21	1.51E–17	1.22E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	2.07E–12	^{138}La	4.68E–15	1.10E–05	5.15E–20	2.57E–16	2.07E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	3.87E–02	^{93m}Nb	8.74E–05	1.10E–05	9.61E–10	4.79E–06	3.87E–02	0.01%
^{94}Nb	[d]		1.66E–04	4.08E–02	^{94}Nb	[e]		9.92E–09	4.95E–05	4.09E–02	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	1.76E–11	^{144}Nd	3.96E–14	1.10E–05	4.36E–19	2.18E–15	1.76E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	4.99E–03	^{59}Ni	1.13E–05	1.10E–05	1.24E–10	6.18E–07	4.99E–03	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	5.68E–01	^{63}Ni	[e]		6.03E–09	3.01E–05	5.68E–01	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	9.77E–08	^{236}Np	2.21E–10	1.10E–05	2.43E–15	1.21E–11	9.77E–08	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	9.35E–03	^{237}Np	[e]		2.71E–11	1.35E–07	9.35E–03	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	1.59E–06	^{238}Np	3.58E–09	1.10E–05	3.94E–14	1.97E–10	1.59E–06	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	4.35E–04	^{239}Np	9.83E–07	1.10E–05	1.08E–11	5.40E–08	4.36E–04	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	1.30E–11	^{240m}Np	2.94E–14	1.10E–05	3.23E–19	1.61E–15	1.30E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	9.39E–07	^{231}Pa	2.12E–09	1.10E–05	2.33E–14	1.16E–10	9.39E–07	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	9.35E–03	^{233}Pa	2.11E–05	1.10E–05	2.32E–10	1.16E–06	9.35E–03	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	4.70E–07	^{234}Pa	1.06E–09	1.10E–05	1.17E–14	5.83E–11	4.70E–07	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	3.62E–04	^{234m}Pa	8.17E–07	1.10E–05	8.98E–12	4.48E–08	3.62E–04	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	2.48E–09	^{209}Pb	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	4.93E–08	^{210}Pb	1.11E–10	1.10E–05	1.23E–15	6.11E–12	4.94E–08	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	5.30E–07	^{211}Pb	1.20E–09	1.10E–05	1.32E–14	6.57E–11	5.31E–07	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	3.18E–05	^{212}Pb	7.18E–08	1.10E–05	7.90E–13	3.94E–09	3.18E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	1.27E–07	^{214}Pb	2.88E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%

Table A-6. (continued).

	Solids				Liquids						Total (Solids+Liquids)
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{107}Pd	4.06E–07	1.8	7.31E–07	1.80E–04	^{107}Pd	4.06E–07	1.10E–05	4.46E–12	2.23E–08	1.80E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	1.77E–04	^{146}Pm	4.00E–07	1.10E–05	4.40E–12	2.19E–08	1.77E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	1.72E–01	^{147}Pm	3.88E–04	1.10E–05	4.27E–09	2.13E–05	1.72E–01	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	4.77E–08	^{210}Po	1.08E–10	1.10E–05	1.18E–15	5.91E–12	4.77E–08	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	1.49E–09	^{211}Po	3.35E–12	1.10E–05	3.69E–17	1.84E–13	1.49E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	2.04E–05	^{212}Po	4.60E–08	1.10E–05	5.06E–13	2.53E–09	2.04E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	2.43E–09	^{213}Po	5.49E–12	1.10E–05	6.04E–17	3.01E–13	2.43E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	1.27E–07	^{214}Po	2.87E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	5.30E–07	^{215}Po	1.20E–09	1.10E–05	1.32E–14	6.57E–11	5.31E–07	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	3.18E–05	^{216}Po	7.18E–08	1.10E–05	7.90E–13	3.94E–09	3.18E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	1.27E–07	^{218}Po	2.88E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	1.03E–07	^{144}Pr	2.33E–10	1.10E–05	2.56E–15	1.28E–11	1.03E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	1.24E–09	^{144m}Pr	2.79E–12	1.10E–05	3.07E–17	1.53E–13	1.24E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	2.89E–06	^{236}Pu	6.52E–09	1.10E–05	7.18E–14	3.58E–10	2.89E–06	0.00%
^{238}Pu	[d]		9.23E–03	2.27E+00	^{238}Pu	[e]		5.37E–08	2.68E–04	2.27E+00	0.48%
^{239}Pu	[d]		2.75E–03	6.77E–01	^{239}Pu	[e]		1.17E–08	5.84E–05	6.77E–01	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	2.68E–01	^{240}Pu	6.06E–04	1.10E–05	6.67E–09	3.33E–05	2.68E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	3.88E+00	^{241}Pu	[e]		2.22E–07	1.11E–03	3.88E+00	0.82%
^{242}Pu	4.43E–07	1.8	7.98E–07	1.96E–04	^{242}Pu	4.43E–07	1.10E–05	4.87E–12	2.43E–08	1.96E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	5.40E–14	^{243}Pu	1.22E–16	1.10E–05	1.34E–21	6.69E–18	5.40E–14	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	1.30E–11	^{244}Pu	2.94E–14	1.10E–05	3.23E–19	1.61E–15	1.30E–11	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	5.30E–07	^{223}Ra	1.20E–09	1.10E–05	1.32E–14	6.57E–11	5.31E–07	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	3.18E–05	^{224}Ra	7.18E–08	1.10E–05	7.90E–13	3.94E–09	3.18E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	2.48E–09	^{225}Ra	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	1.27E–07	^{226}Ra	2.88E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%

Table A-6. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{228}Ra	1.73E–14	1.8	3.11E–14	7.65E–12	^{228}Ra	1.73E–14	1.10E–05	1.90E–19	9.48E–16	7.65E–12	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	3.16E–07	^{87}Rb	7.15E–10	1.10E–05	7.86E–15	3.92E–11	3.17E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	1.09E–06	^{102}Rh	2.46E–09	1.10E–05	2.71E–14	1.35E–10	1.09E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	3.40E–06	^{106}Rh	7.69E–09	1.10E–05	8.45E–14	4.22E–10	3.40E–06	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	5.30E–07	^{219}Rn	1.20E–09	1.10E–05	1.32E–14	6.57E–11	5.31E–07	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	3.18E–05	^{220}Rn	7.18E–08	1.10E–05	7.90E–13	3.94E–09	3.18E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	1.27E–07	^{222}Rn	2.88E–10	1.10E–05	3.16E–15	1.58E–11	1.27E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	3.40E–06	^{106}Ru	7.69E–09	1.10E–05	8.45E–14	4.22E–10	3.40E–06	0.00%
^{125}Sb	[d]		5.55E–04	1.37E–01	^{125}Sb	[e]		1.14E–08	5.69E–05	1.37E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	6.27E–04	^{126}Sb	1.41E–06	1.10E–05	1.56E–11	7.76E–08	6.27E–04	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	4.48E–03	^{126m}Sb	1.01E–05	1.10E–05	1.11E–10	5.55E–07	4.48E–03	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	4.75E–03	^{79}Se	1.07E–05	1.10E–05	1.18E–10	5.89E–07	4.75E–03	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	2.06E–09	^{146}Sm	4.65E–12	1.10E–05	5.11E–17	2.55E–13	2.06E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	8.03E–08	^{147}Sm	1.81E–10	1.10E–05	1.99E–15	9.95E–12	8.03E–08	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	4.12E–13	^{148}Sm	9.30E–16	1.10E–05	1.02E–20	5.10E–17	4.12E–13	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	3.66E–14	^{149}Sm	8.26E–17	1.10E–05	9.08E–22	4.53E–18	3.66E–14	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	3.41E+00	^{151}Sm	7.71E–03	1.10E–05	8.48E–08	4.23E–04	3.41E+00	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	4.23E–12	^{119m}Sn	9.55E–15	1.10E–05	1.05E–19	5.24E–16	4.23E–12	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	2.54E–02	^{121m}Sn	5.74E–05	1.10E–05	6.31E–10	3.15E–06	2.54E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	4.48E–03	^{126}Sn	1.01E–05	1.10E–05	1.11E–10	5.55E–07	4.48E–03	0.00%
^{90}Sr	[d]		1.87E–02	4.60E+00	^{90}Sr	[e]		3.72E–07	1.86E–03	4.60E+00	0.97%
^{98}Tc	6.34E–11	1.8	1.14E–10	2.81E–08	^{98}Tc	6.34E–11	1.10E–05	6.98E–16	3.48E–12	2.81E–08	0.00%
^{99}Tc	[d]		6.17E–04	1.52E–01	^{99}Tc	[e]		2.88E–09	1.44E–05	1.52E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	2.27E–14	^{123}Te	5.12E–17	1.10E–05	5.64E–22	2.81E–18	2.27E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	3.35E–02	^{125m}Te	1.43E–05	1.10E–05	1.58E–10	7.86E–07	3.35E–02	0.01%

Table A-6. (continued).

Solids					Liquids						Total (Solids+Liquids)
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{227}Th	1.18E–09	1.8	2.13E–09	5.23E–07	^{227}Th	1.18E–09	1.10E–05	1.30E–14	6.48E–11	5.23E–07	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	3.17E–05	^{228}Th	7.16E–08	1.10E–05	7.88E–13	3.93E–09	3.17E–05	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	2.48E–09	^{229}Th	5.61E–12	1.10E–05	6.17E–17	3.08E–13	2.48E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	1.04E–05	^{230}Th	2.35E–08	1.10E–05	2.59E–13	1.29E–09	1.04E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	3.51E–04	^{231}Th	7.93E–07	1.10E–05	8.72E–12	4.35E–08	3.51E–04	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	8.19E–12	^{232}Th	1.85E–14	1.10E–05	2.04E–19	1.02E–15	8.20E–12	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	3.62E–04	^{234}Th	8.17E–07	1.10E–05	8.98E–12	4.48E–08	3.62E–04	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	5.29E–07	^{207}Tl	1.19E–09	1.10E–05	1.31E–14	6.56E–11	5.29E–07	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	1.14E–05	^{208}Tl	2.58E–08	1.10E–05	2.84E–13	1.42E–09	1.14E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	5.37E–11	^{209}Tl	1.21E–13	1.10E–05	1.33E–18	6.65E–15	5.37E–11	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	2.12E–13	^{171}Tm	4.80E–16	1.10E–05	5.28E–21	2.63E–17	2.12E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	3.06E–05	^{232}U	6.91E–08	1.10E–05	7.60E–13	3.79E–09	3.06E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	7.86E–07	^{233}U	1.77E–09	1.10E–05	1.95E–14	9.74E–11	7.86E–07	0.00%
^{234}U	[d]		2.98E–06	7.33E–04	^{234}U	[e]		4.45E–11	2.22E–07	7.34E–04	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	3.51E–04	^{235}U	7.93E–07	1.10E–05	8.72E–12	4.35E–08	3.51E–04	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	8.19E–04	^{236}U	1.85E–06	1.10E–05	2.03E–11	1.01E–07	8.19E–04	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	9.51E–05	^{237}U	2.15E–07	1.10E–05	2.36E–12	1.18E–08	9.51E–05	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	3.62E–04	^{238}U	8.17E–07	1.10E–05	8.98E–12	4.48E–08	3.62E–04	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	1.30E–11	^{240}U	2.94E–14	1.10E–05	3.23E–19	1.61E–15	1.30E–11	0.00%
^{90}Y	[d]		1.87E–02	4.60E+00	^{90}Y	8.88E–01	1.10E–05	9.77E–06	4.88E–02	4.65E+00	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	4.68E–02	^{93}Zr	1.06E–04	1.10E–05	1.16E–09	5.80E–06	4.68E–02	0.01%
			Total	475.24				Total	0.12	475.4	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-181.

Table A-7. Post-decontamination estimated inventory for Tank WM-182.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	1.25E–08	^{225}Ac	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	2.66E–06	^{227}Ac	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.66E–06	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	3.85E–11	^{228}Ac	1.73E–14	2.23E–04	3.85E–18	1.92E–14	3.85E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	6.35E–10	^{108}Ag	2.85E–13	2.23E–04	6.35E–17	3.17E–13	6.35E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	7.13E–09	^{108m}Ag	3.20E–12	2.23E–04	7.14E–16	3.56E–12	7.14E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	5.30E–14	^{109m}Ag	2.38E–17	2.23E–04	5.30E–21	2.65E–17	5.30E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	1.99E–15	^{110}Ag	8.93E–19	2.23E–04	1.99E–22	9.93E–19	1.99E–15	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	1.50E–13	^{110m}Ag	6.71E–17	2.23E–04	1.50E–20	7.47E–17	1.50E–13	0.00%
^{241}Am	[d]		3.40E–04	4.21E–01	^{241}Am	[e]		1.06E–07	5.30E–04	4.21E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	1.59E–03	^{242}Am	7.13E–07	2.23E–04	1.59E–10	7.94E–07	1.59E–03	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	1.60E–03	^{242m}Am	7.17E–07	2.23E–04	1.60E–10	7.98E–07	1.60E–03	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	2.19E–03	^{243}Am	9.83E–07	2.23E–04	2.19E–10	1.09E–06	2.19E–03	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	1.25E–08	^{217}At	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{137m}Ba	[d]		9.20E–01	1.14E+03	^{137m}Ba	[e]		2.23E-04	1.11E+00	1.14E+03	47.60%
^{10}Be	7.56E–11	1.8	1.36E–10	1.69E–07	^{10}Be	7.56E–11	2.23E–04	1.69E–14	8.41E–11	1.69E–07	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	2.48E–07	^{210}Bi	1.11E–10	2.23E–04	2.49E–14	1.24E–10	2.48E–07	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	1.21E–20	^{210m}Bi	5.41E–24	2.23E–04	1.21E–27	6.02E–24	1.21E–20	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	2.67E–06	^{211}Bi	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.67E–06	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	1.60E–04	^{212}Bi	7.18E–08	2.23E–04	1.60E–11	7.99E–08	1.60E–04	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	1.25E–08	^{213}Bi	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	6.41E–07	^{214}Bi	2.88E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	4.90E–06	^{14}C	[e]		1.08E–11	5.39E–08	4.96E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	5.30E–14	^{109}Cd	2.38E–17	2.23E–04	5.30E–21	2.65E–17	5.30E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	1.29E–01	^{113m}Cd	5.78E–05	2.23E–04	1.29E–08	6.43E–05	1.29E–01	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	1.63E–06	^{142}Ce	7.31E–10	2.23E–04	1.63E–13	8.14E–10	1.63E–06	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	5.18E–07	^{144}Ce	2.33E–10	2.23E–04	5.19E–14	2.59E–10	5.19E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	1.61E–12	^{249}Cf	7.21E–16	2.23E–04	1.61E–19	8.02E–16	1.61E–12	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	8.31E–13	^{250}Cf	3.73E–16	2.23E–04	8.32E–20	4.15E–16	8.31E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	2.55E–14	^{251}Cf	1.14E–17	2.23E–04	2.55E–21	1.27E–17	2.55E–14	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	1.08E–15	^{252}Cf	4.84E–19	2.23E–04	1.08E–22	5.39E–19	1.08E–15	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	1.32E–03	^{242}Cm	5.91E–07	2.23E–04	1.32E–10	6.58E–07	1.32E–03	0.00%

Table A-7. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{243}Cm	1.29E–07	1.8	2.31E–07	2.86E–04	^{243}Cm	1.29E–07	2.23E–04	2.87E–11	1.43E–07	2.87E–04	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	1.57E–02	^{244}Cm	7.03E–06	2.23E–04	1.57E–09	7.82E–06	1.57E–02	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	3.74E–06	^{245}Cm	1.68E–09	2.23E–04	3.75E–13	1.87E–09	3.75E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	2.45E–07	^{246}Cm	1.10E–10	2.23E–04	2.46E–14	1.23E–10	2.46E–07	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	2.72E–13	^{247}Cm	1.22E–16	2.23E–04	2.72E–20	1.36E–16	2.72E–13	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	2.87E–13	^{248}Cm	1.29E–16	2.23E–04	2.87E–20	1.43E–16	2.87E–13	0.00%
^{60}Co	[d]		5.02E–05	6.21E–02	^{60}Co	3.70E–04	2.23E–04	8.26E–08	4.12E–04	6.26E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	6.76E–02	^{134}Cs	3.03E–05	2.23E–04	6.76E–09	3.37E–05	6.76E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	3.20E–02	^{135}Cs	1.44E–05	2.23E–04	3.21E–09	1.60E–05	3.21E–02	0.00%
^{137}Cs	[d]		9.20E–01	1.14E+03	^{137}Cs	[e]		2.23E–04	1.11E+00	1.14E+03	47.65%
^{150}Eu	2.96E–10	1.8	5.34E–10	6.61E–07	^{150}Eu	2.96E–10	2.23E–04	6.61E–14	3.30E–10	6.61E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	8.73E–02	^{152}Eu	3.92E–05	2.23E–04	8.74E–09	4.36E–05	8.74E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	3.89E+00	^{154}Eu	[e]		6.40E–08	3.19E–04	3.89E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	1.06E+00	^{155}Eu	4.74E–04	2.23E–04	1.06E–07	5.27E–04	1.06E+00	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	1.09E+00	^{55}Fe	4.88E–04	2.23E–04	1.09E–07	5.43E–04	1.09E+00	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	1.25E–08	^{221}Fr	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	3.68E–08	^{223}Fr	1.65E–11	2.23E–04	3.68E–15	1.84E–11	3.68E–08	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	7.97E–14	^{152}Gd	3.58E–17	2.23E–04	7.98E–21	3.98E–17	7.98E–14	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	9.26E–16	^{153}Gd	4.15E–19	2.23E–04	9.26E–23	4.62E–19	9.26E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	7.17E–01	^{3}H	[e]		3.33E–09	1.66E–05	7.17E–01	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	2.51E–06	^{166m}Ho	1.13E–09	2.23E–04	2.51E–13	1.25E–09	2.51E–06	0.00%
^{129}I	[d]		6.24E–07	7.73E–04	^{129}I	[e]		2.25E–10	1.12E–06	7.74E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	6.13E–13	^{115}In	2.75E–16	2.23E–04	6.13E–20	3.06E–16	6.13E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	1.04E–11	^{138}La	4.68E–15	2.23E–04	1.04E–18	5.21E–15	1.04E–11	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	1.95E–01	^{93m}Nb	8.74E–05	2.23E–04	1.95E–08	9.72E–05	1.95E–01	0.01%
^{94}Nb	[d]		1.66E–04	2.06E–01	^{94}Nb	3.62E–05	2.23E–04	8.07E–09	4.03E–05	2.06E–01	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	8.83E–11	^{144}Nd	3.96E–14	2.23E–04	8.84E–18	4.41E–14	8.84E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	2.51E–02	^{59}Ni	1.13E–05	2.23E–04	2.51E–09	1.25E–05	2.51E–02	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	2.86E+00	^{63}Ni	1.28E–03	2.23E–04	2.86E–07	1.43E–03	2.86E+00	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	4.91E–07	^{236}Np	2.21E–10	2.23E–04	4.92E–14	2.45E–10	4.92E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	4.70E–02	^{237}Np	[e]		5.44E–11	2.71E–07	4.70E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	7.99E–06	^{238}Np	3.58E–09	2.23E–04	7.99E–13	3.99E–09	7.99E–06	0.00%

Table A-7. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{239}Np	9.83E–07	1.8	1.77E–06	2.19E–03	^{239}Np	9.83E–07	2.23E–04	2.19E–10	1.09E–06	2.19E–03	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	6.54E–11	^{240m}Np	2.94E–14	2.23E–04	6.55E–18	3.27E–14	6.55E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	4.73E–06	^{231}Pa	2.12E–09	2.23E–04	4.73E–13	2.36E–09	4.73E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	4.70E–02	^{233}Pa	2.11E–05	2.23E–04	4.71E–09	2.35E–05	4.70E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	2.37E–06	^{234}Pa	1.06E–09	2.23E–04	2.37E–13	1.18E–09	2.37E–06	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	1.82E–03	^{234m}Pa	8.17E–07	2.23E–04	1.82E–10	9.09E–07	1.82E–03	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	1.25E–08	^{209}Pb	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	2.48E–07	^{210}Pb	1.11E–10	2.23E–04	2.48E–14	1.24E–10	2.48E–07	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	2.67E–06	^{211}Pb	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.67E–06	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	1.60E–04	^{212}Pb	7.18E–08	2.23E–04	1.60E–11	7.99E–08	1.60E–04	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	6.41E–07	^{214}Pb	2.88E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	9.05E–04	^{107}Pd	4.06E–07	2.23E–04	9.05E–11	4.52E–07	9.05E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	8.91E–04	^{146}Pm	4.00E–07	2.23E–04	8.92E–11	4.45E–07	8.92E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	8.65E–01	^{147}Pm	3.88E–04	2.23E–04	8.66E–08	4.32E–04	8.66E–01	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	2.40E–07	^{210}Po	1.08E–10	2.23E–04	2.40E–14	1.20E–10	2.40E–07	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	7.47E–09	^{211}Po	3.35E–12	2.23E–04	7.48E–16	3.73E–12	7.48E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	1.03E–04	^{212}Po	4.60E–08	2.23E–04	1.03E–11	5.12E–08	1.03E–04	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	1.22E–08	^{213}Po	5.49E–12	2.23E–04	1.22E–15	6.11E–12	1.22E–08	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	6.41E–07	^{214}Po	2.87E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	2.67E–06	^{215}Po	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.67E–06	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	1.60E–04	^{216}Po	7.18E–08	2.23E–04	1.60E–11	7.99E–08	1.60E–04	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	6.41E–07	^{218}Po	2.88E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	5.18E–07	^{144}Pr	2.33E–10	2.23E–04	5.19E–14	2.59E–10	5.19E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	6.22E–09	^{144m}Pr	2.79E–12	2.23E–04	6.23E–16	3.11E–12	6.22E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	1.45E–05	^{236}Pu	6.52E–09	2.23E–04	1.45E–12	7.26E–09	1.45E–05	0.00%
^{238}Pu	[d]		9.23E–03	1.14E+01	^{238}Pu	[e]		4.96E–07	2.47E–03	1.14E+01	0.48%
^{239}Pu	[d]		2.75E–03	3.40E+00	^{239}Pu	[e]		4.90E–08	2.44E–04	3.40E+00	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	1.35E+00	^{240}Pu	6.06E–04	2.23E–04	1.35E–07	6.74E–04	1.35E+00	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	1.95E+01	^{241}Pu	[e]		1.18E–07	5.89E–04	1.95E+01	0.81%
^{242}Pu	4.43E–07	1.8	7.98E–07	9.87E–04	^{242}Pu	4.43E–07	2.23E–04	9.88E–11	4.93E–07	9.88E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	2.72E–13	^{243}Pu	1.22E–16	2.23E–04	2.72E–20	1.36E–16	2.72E–13	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	6.55E–11	^{244}Pu	2.94E–14	2.23E–04	6.56E–18	3.27E–14	6.55E–11	0.00%

Table A-7. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{223}Ra	1.20E–09	1.8	2.16E–09	2.67E–06	^{223}Ra	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.67E–06	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	1.60E–04	^{224}Ra	7.18E–08	2.23E–04	1.60E–11	7.99E–08	1.60E–04	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	1.25E–08	^{225}Ra	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	6.41E–07	^{226}Ra	2.88E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	3.85E–11	^{228}Ra	1.73E–14	2.23E–04	3.85E–18	1.92E–14	3.85E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	1.59E–06	^{87}Rb	7.15E–10	2.23E–04	1.59E–13	7.95E–10	1.59E–06	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	5.49E–06	^{102}Rh	2.46E–09	2.23E–04	5.50E–13	2.74E–09	5.49E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	1.71E–05	^{106}Rh	7.69E–09	2.23E–04	1.71E–12	8.55E–09	1.71E–05	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	2.67E–06	^{219}Rn	1.20E–09	2.23E–04	2.67E–13	1.33E–09	2.67E–06	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	1.60E–04	^{220}Rn	7.18E–08	2.23E–04	1.60E–11	7.99E–08	1.60E–04	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	6.41E–07	^{222}Rn	2.88E–10	2.23E–04	6.41E–14	3.20E–10	6.41E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	1.71E–05	^{106}Ru	7.69E–09	2.23E–04	1.71E–12	8.55E–09	1.71E–05	0.00%
^{125}Sb	[d]		5.55E–04	6.87E–01	^{125}Sb	[e]		3.53E–07	1.76E–03	6.89E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	3.15E–03	^{126}Sb	1.41E–06	2.23E–04	3.15E–10	1.57E–06	3.15E–03	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	2.25E–02	^{126m}Sb	1.01E–05	2.23E–04	2.25E–09	1.12E–05	2.25E–02	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	2.39E–02	^{79}Se	1.07E–05	2.23E–04	2.39E–09	1.19E–05	2.39E–02	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	1.04E–08	^{146}Sm	4.65E–12	2.23E–04	1.04E–15	5.17E–12	1.04E–08	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	4.04E–07	^{147}Sm	1.81E–10	2.23E–04	4.04E–14	2.02E–10	4.04E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	2.07E–12	^{148}Sm	9.30E–16	2.23E–04	2.07E–19	1.03E–15	2.07E–12	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	1.84E–13	^{149}Sm	8.26E–17	2.23E–04	1.84E–20	9.19E–17	1.84E–13	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	1.72E+01	^{151}Sm	7.71E–03	2.23E–04	1.72E–06	8.57E–03	1.72E+01	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	2.13E–11	^{119m}Sn	9.55E–15	2.23E–04	2.13E–18	1.06E–14	2.13E–11	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	1.28E–01	^{121m}Sn	5.74E–05	2.23E–04	1.28E–08	6.38E–05	1.28E–01	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	2.25E–02	^{126}Sn	1.01E–05	2.23E–04	2.25E–09	1.12E–05	2.25E–02	0.00%
^{90}Sr	[d]		1.87E–02	2.32E+01	^{90}Sr	[e]		4.83E–05	2.41E–01	2.34E+01	0.98%
^{98}Tc	6.34E–11	1.8	1.14E–10	1.41E–07	^{98}Tc	6.34E–11	2.23E–04	1.41E–14	7.06E–11	1.41E–07	0.00%
^{99}Tc	[d]		6.17E–04	7.64E–01	^{99}Tc	[e]		9.11E–09	4.54E–05	7.64E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	1.14E–13	^{123}Te	5.12E–17	2.23E–04	1.14E–20	5.70E–17	1.14E–13	0.00%
^{125m}Te	1.43E–05		1.36E–04	1.68E–01	^{125m}Te	1.43E–05	2.23E–04	3.19E–09	1.59E–05	1.68E–01	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	2.63E–06	^{227}Th	1.18E–09	2.23E–04	2.63E–13	1.31E–09	2.63E–06	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	1.60E–04	^{228}Th	7.16E–08	2.23E–04	1.60E–11	7.97E–08	1.60E–04	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	1.25E–08	^{229}Th	5.61E–12	2.23E–04	1.25E–15	6.24E–12	1.25E–08	0.00%

Table A-7. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{230}Th	2.35E–08	1.8	4.24E–08	5.24E–05	^{230}Th	2.35E–08	2.23E–04	5.25E–12	2.62E–08	5.25E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	1.77E–03	^{231}Th	7.93E–07	2.23E–04	1.77E–10	8.82E–07	1.77E–03	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	4.12E–11	^{232}Th	1.85E–14	2.23E–04	4.13E–18	2.06E–14	4.12E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	1.82E–03	^{234}Th	8.17E–07	2.23E–04	1.82E–10	9.09E–07	1.82E–03	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	2.66E–06	^{207}Tl	1.19E–09	2.23E–04	2.66E–13	1.33E–09	2.66E–06	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	5.75E–05	^{208}Tl	2.58E–08	2.23E–04	5.76E–12	2.87E–08	5.75E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	2.70E–10	^{209}Tl	1.21E–13	2.23E–04	2.70E–17	1.35E–13	2.70E–10	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	1.07E–12	^{171}Tm	4.80E–16	2.23E–04	1.07E–19	5.34E–16	1.07E–12	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	1.54E–04	^{232}U	6.91E–08	2.23E–04	1.54E–11	7.69E–08	1.54E–04	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	3.95E–06	^{233}U	1.77E–09	2.23E–04	3.96E–13	1.97E–09	3.96E–06	0.00%
^{234}U	[d]		2.98E–06	3.69E–03	^{234}U	[e]		6.18E–10	3.08E–06	3.69E–03	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	1.77E–03	^{235}U	7.93E–07	2.23E–04	1.77E–10	8.82E–07	1.77E–03	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	4.12E–03	^{236}U	1.85E–06	2.23E–04	4.12E–10	2.06E–06	4.12E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	4.78E–04	^{237}U	2.15E–07	2.23E–04	4.79E–11	2.39E–07	4.79E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	1.82E–03	^{238}U	8.17E–07	2.23E–04	1.82E–10	9.09E–07	1.82E–03	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	6.54E–11	^{240}U	2.94E–14	2.23E–04	6.55E–18	3.27E–14	6.55E–11	0.00%
^{90}Y	[d]		1.87E–02	2.32E+01	^{90}Y	[e]		4.83E–05	2.41E–01	2.34E+01	0.98%
^{93}Zr	1.06E–04	1.8	1.90E–04	2.36E–01	^{93}Zr	1.06E–04	2.23E–04	2.36E–08	1.18E–04	2.36E–01	0.01%
			Total	2,391				Total	3	2,394	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-182.

Table A-8. Post-decontamination estimated inventory for Tank WM-183.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	7.08E–09	^{225}Ac	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	1.51E–06	^{227}Ac	1.20E–09	8.38E–05	1.00E–13	5.00E–10	1.51E–06	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	2.18E–11	^{228}Ac	1.73E–14	8.38E–05	1.45E–18	7.22E–15	2.18E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	3.60E–10	^{108}Ag	2.85E–13	8.38E–05	2.39E–17	1.19E–13	3.60E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	4.04E–09	^{108m}Ag	3.20E–12	8.38E–05	2.68E–16	1.34E–12	4.05E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	3.00E–14	^{109m}Ag	2.38E–17	8.38E–05	1.99E–21	9.94E–18	3.00E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	1.13E–15	^{110}Ag	8.93E–19	8.38E–05	7.48E–23	3.73E–19	1.13E–15	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	8.48E–14	^{110m}Ag	6.71E–17	8.38E–05	5.63E–21	2.81E–17	8.48E–14	0.00%
^{241}Am	[d]		3.40E–04	2.39E–01	^{241}Am	[e]		1.94E–07	9.68E–04	2.40E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	9.01E–04	^{242}Am	7.13E–07	8.38E–05	5.98E–11	2.98E–07	9.01E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	9.06E–04	^{242m}Am	7.17E–07	8.38E–05	6.01E–11	3.00E–07	9.06E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	1.24E–03	^{243}Am	9.83E–07	8.38E–05	8.24E–11	4.11E–07	1.24E–03	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	7.08E–09	^{217}At	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{137m}Ba	[d]		9.20E–01	6.46E+02	^{137m}Ba	[e]		8.38E–05	4.18E–01	6.46E+02	47.39%
^{10}Be	7.56E–11	1.8	1.36E–10	9.55E–08	^{10}Be	7.56E–11	8.38E–05	6.34E–15	3.16E–11	9.56E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	1.41E–07	^{210}Bi	1.11E–10	8.38E–05	9.34E–15	4.66E–11	1.41E–07	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	6.84E–21	^{210m}Bi	5.41E–24	8.38E–05	4.54E–28	2.26E–24	6.84E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	1.51E–06	^{211}Bi	1.20E–09	8.38E–05	1.00E–13	5.01E–10	1.51E–06	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	9.07E–05	^{212}Bi	7.18E–08	8.38E–05	6.02E–12	3.00E–08	9.08E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	7.08E–09	^{213}Bi	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	3.63E–07	^{214}Bi	2.88E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	2.78E–06	^{14}C	[e]		1.72E–11	8.58E–08	2.86E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	3.00E–14	^{109}Cd	2.38E–17	8.38E–05	1.99E–21	9.94E–18	3.00E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	7.30E–02	^{113m}Cd	5.78E–05	8.38E–05	4.84E–09	2.42E–05	7.30E–02	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	9.24E–07	^{142}Ce	7.31E–10	8.38E–05	6.13E–14	3.06E–10	9.24E–07	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	2.94E–07	^{144}Ce	2.33E–10	8.38E–05	1.95E–14	9.73E–11	2.94E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	9.10E–13	^{249}Cf	7.21E–16	8.38E–05	6.04E–20	3.01E–16	9.10E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	4.71E–13	^{250}Cf	3.73E–16	8.38E–05	3.12E–20	1.56E–16	4.71E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	1.44E–14	^{251}Cf	1.14E–17	8.38E–05	9.57E–22	4.78E–18	1.44E–14	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	6.12E–16	^{252}Cf	4.84E–19	8.38E–05	4.06E–23	2.02E–19	6.12E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	7.47E–04	^{242}Cm	5.91E–07	8.38E–05	4.96E–11	2.47E–07	7.47E–04	0.00%

Table A-8. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{243}Cm	1.29E–07	1.8	2.31E–07	1.62E–04	^{243}Cm	1.29E–07	8.38E–05	1.08E–11	5.37E–08	1.62E–04	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	8.88E–03	^{244}Cm	[e]		3.25E–09	1.62E–05	8.90E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	2.12E–06	^{245}Cm	1.68E–09	8.38E–05	1.41E–13	7.02E–10	2.12E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	1.39E–07	^{246}Cm	1.10E–10	8.38E–05	9.23E–15	4.61E–11	1.39E–07	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	1.54E–13	^{247}Cm	1.22E–16	8.38E–05	1.02E–20	5.10E–17	1.54E–13	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	1.63E–13	^{248}Cm	1.29E–16	8.38E–05	1.08E–20	5.38E–17	1.63E–13	0.00%
^{60}Co	[d]		5.02E–05	3.52E–02	^{60}Co	[e]		2.66E–09	1.33E–05	3.52E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	3.83E–02	^{134}Cs	3.03E–05	8.38E–05	2.54E–09	1.27E–05	3.83E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	1.82E–02	^{135}Cs	1.44E–05	8.38E–05	1.21E–09	6.01E–06	1.82E–02	0.00%
^{137}Cs	[d]		9.20E–01	6.46E+02	^{137}Cs	[e]		8.38E–05	4.18E–01	6.46E+02	47.39%
^{150}Eu	2.96E–10	1.8	5.34E–10	3.74E–07	^{150}Eu	2.96E–10	8.38E–05	2.48E–14	1.24E–10	3.75E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	4.95E–02	^{152}Eu	3.92E–05	8.38E–05	3.28E–09	1.64E–05	4.95E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	2.21E+00	^{154}Eu	[e]		6.35E–09	3.17E–05	2.21E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	5.98E–01	^{155}Eu	4.74E–04	8.38E–05	3.97E–08	1.98E–04	5.98E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	6.16E–01	^{55}Fe	4.88E–04	8.38E–05	4.09E–08	2.04E–04	6.17E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	7.08E–09	^{221}Fr	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	2.08E–08	^{223}Fr	1.65E–11	8.38E–05	1.38E–15	6.90E–12	2.08E–08	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	4.52E–14	^{152}Gd	3.58E–17	8.38E–05	3.00E–21	1.50E–17	4.52E–14	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	5.25E–16	^{153}Gd	4.15E–19	8.38E–05	3.48E–23	1.74E–19	5.25E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	4.06E–01	^{3}H	[e]		3.87E–07	1.93E–03	4.08E–01	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	1.42E–06	^{166m}Ho	1.13E–09	8.38E–05	9.44E–14	4.71E–10	1.42E–06	0.00%
^{129}I	[d]		6.24E–07	4.38E–04	^{129}I	[e]		1.53E–09	7.63E–06	4.46E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	3.47E–13	^{115}In	2.75E–16	8.38E–05	2.30E–20	1.15E–16	3.48E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	5.92E–12	^{138}La	4.68E–15	8.38E–05	3.93E–19	1.96E–15	5.92E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	1.10E–01	^{93m}Nb	8.74E–05	8.38E–05	7.32E–09	3.65E–05	1.10E–01	0.01%
^{94}Nb	[d]		1.66E–04	1.16E–01	^{94}Nb	[e]		1.09E–08	5.44E–05	1.17E–01	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	5.01E–11	^{144}Nd	3.96E–14	8.38E–05	3.32E–18	1.66E–14	5.01E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	1.42E–02	^{59}Ni	1.13E–05	8.38E–05	9.44E–10	4.71E–06	1.42E–02	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	1.62E+00	^{63}Ni	[e]		2.84E–07	1.42E–03	1.62E+00	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	2.79E–07	^{236}Np	2.21E–10	8.38E–05	1.85E–14	9.22E–11	2.79E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	2.67E–02	^{237}Np	[e]		3.12E–09	1.56E–05	2.67E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	4.53E–06	^{238}Np	3.58E–09	8.38E–05	3.00E–13	1.50E–09	4.53E–06	0.00%

Table A-8. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{239}Np	9.83E–07	1.8	1.77E–06	1.24E–03	^{239}Np	9.83E–07	8.38E–05	8.24E–11	4.11E–07	1.24E–03	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	3.71E–11	^{240m}Np	2.94E–14	8.38E–05	2.46E–18	1.23E–14	3.71E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	2.68E–06	^{231}Pa	2.12E–09	8.38E–05	1.78E–13	8.87E–10	2.68E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	2.67E–02	^{233}Pa	2.11E–05	8.38E–05	1.77E–09	8.82E–06	2.67E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	1.34E–06	^{234}Pa	1.06E–09	8.38E–05	8.89E–14	4.44E–10	1.34E–06	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	1.03E–03	^{234m}Pa	8.17E–07	8.38E–05	6.84E–11	3.41E–07	1.03E–03	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	7.08E–09	^{209}Pb	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	1.41E–07	^{210}Pb	1.11E–10	8.38E–05	9.34E–15	4.66E–11	1.41E–07	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	1.51E–06	^{211}Pb	1.20E–09	8.38E–05	1.00E–13	5.01E–10	1.51E–06	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	9.07E–05	^{212}Pb	7.18E–08	8.38E–05	6.02E–12	3.00E–08	9.08E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	3.63E–07	^{214}Pb	2.88E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	5.13E–04	^{107}Pd	4.06E–07	8.38E–05	3.40E–11	1.70E–07	5.13E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	5.05E–04	^{146}Pm	4.00E–07	8.38E–05	3.35E–11	1.67E–07	5.05E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	4.91E–01	^{147}Pm	3.88E–04	8.38E–05	3.25E–08	1.62E–04	4.91E–01	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	1.36E–07	^{210}Po	1.08E–10	8.38E–05	9.02E–15	4.50E–11	1.36E–07	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	4.24E–09	^{211}Po	3.35E–12	8.38E–05	2.81E–16	1.40E–12	4.24E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	5.81E–05	^{212}Po	4.60E–08	8.38E–05	3.86E–12	1.92E–08	5.82E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	6.93E–09	^{213}Po	5.49E–12	8.38E–05	4.60E–16	2.29E–12	6.93E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	3.63E–07	^{214}Po	2.87E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	1.51E–06	^{215}Po	1.20E–09	8.38E–05	1.00E–13	5.01E–10	1.51E–06	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	9.07E–05	^{216}Po	7.18E–08	8.38E–05	6.02E–12	3.00E–08	9.08E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	3.63E–07	^{218}Po	2.88E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	2.94E–07	^{144}Pr	2.33E–10	8.38E–05	1.95E–14	9.73E–11	2.94E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	3.53E–09	^{144m}Pr	2.79E–12	8.38E–05	2.34E–16	1.17E–12	3.53E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	8.24E–06	^{236}Pu	6.52E–09	8.38E–05	5.47E–13	2.73E–09	8.24E–06	0.00%
^{238}Pu	[d]		9.23E–03	6.48E+00	^{238}Pu	[e]		9.59E–07	4.78E–03	6.48E+00	0.48%
^{239}Pu	[d]		2.75E–03	1.93E+00	^{239}Pu	[e]		3.37E–07	1.68E–03	1.93E+00	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	7.65E–01	^{240}Pu	6.06E–04	8.38E–05	5.08E–08	2.53E–04	7.66E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	1.11E+01	^{241}Pu	[e]		8.75E–08	4.37E–04	1.11E+01	0.81%
^{242}Pu	4.43E–07	1.8	7.98E–07	5.60E–04	^{242}Pu	4.43E–07	8.38E–05	3.71E–11	1.85E–07	5.60E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	1.54E–13	^{243}Pu	1.22E–16	8.38E–05	1.02E–20	5.10E–17	1.54E–13	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	3.71E–11	^{244}Pu	2.94E–14	8.38E–05	2.46E–18	1.23E–14	3.71E–11	0.00%

Table A-8. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{223}Ra	1.20E–09	1.8	2.16E–09	1.51E–06	^{223}Ra	1.20E–09	8.38E–05	1.00E–13	5.01E–10	1.51E–06	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	9.07E–05	^{224}Ra	7.18E–08	8.38E–05	6.02E–12	3.00E–08	9.08E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	7.08E–09	^{225}Ra	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	3.63E–07	^{226}Ra	2.88E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	2.18E–11	^{228}Ra	1.73E–14	8.38E–05	1.45E–18	7.22E–15	2.18E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	9.03E–07	^{87}Rb	7.15E–10	8.38E–05	5.99E–14	2.99E–10	9.03E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	3.11E–06	^{102}Rh	2.46E–09	8.38E–05	2.06E–13	1.03E–09	3.11E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	9.71E–06	^{106}Rh	7.69E–09	8.38E–05	6.44E–13	3.21E–09	9.71E–06	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	1.51E–06	^{219}Rn	1.20E–09	8.38E–05	1.00E–13	5.01E–10	1.51E–06	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	9.07E–05	^{220}Rn	7.18E–08	8.38E–05	6.02E–12	3.00E–08	9.08E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	3.63E–07	^{222}Rn	2.88E–10	8.38E–05	2.41E–14	1.20E–10	3.63E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	9.71E–06	^{106}Ru	7.69E–09	8.38E–05	6.44E–13	3.21E–09	9.71E–06	0.00%
^{125}Sb	[d]		5.55E–04	3.89E–01	^{125}Sb	[e]		3.34E–08	1.67E–04	3.90E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	1.79E–03	^{126}Sb	1.41E–06	8.38E–05	1.19E–10	5.91E–07	1.79E–03	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	1.28E–02	^{126m}Sb	1.01E–05	8.38E–05	8.47E–10	4.22E–06	1.28E–02	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	1.36E–02	^{79}Se	1.07E–05	8.38E–05	8.99E–10	4.49E–06	1.36E–02	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	5.87E–09	^{146}Sm	4.65E–12	8.38E–05	3.89E–16	1.94E–12	5.87E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	2.29E–07	^{147}Sm	1.81E–10	8.38E–05	1.52E–14	7.58E–11	2.29E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	1.17E–12	^{148}Sm	9.30E–16	8.38E–05	7.79E–20	3.89E–16	1.18E–12	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	1.04E–13	^{149}Sm	8.26E–17	8.38E–05	6.92E–21	3.45E–17	1.04E–13	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	9.73E+00	^{151}Sm	7.71E–03	8.38E–05	6.46E–07	3.22E–03	9.74E+00	0.71%
^{119m}Sn	9.55E–15	1.8	1.72E–14	1.21E–11	^{119m}Sn	9.55E–15	8.38E–05	8.00E–19	3.99E–15	1.21E–11	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	7.25E–02	^{121m}Sn	5.74E–05	8.38E–05	4.81E–09	2.40E–05	7.25E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	1.28E–02	^{126}Sn	1.01E–05	8.38E–05	8.47E–10	4.22E–06	1.28E–02	0.00%
^{90}Sr	[d]		1.87E–02	1.31E+01	^{90}Sr	[e]		7.00E–04	3.49E+00	1.66E+01	1.22%
^{98}Tc	6.34E–11	1.8	1.14E–10	8.01E–08	^{98}Tc	6.34E–11	8.38E–05	5.32E–15	2.65E–11	8.02E–08	0.00%
^{99}Tc	[d]		6.17E–04	4.33E–01	^{99}Tc	[e]		1.75E–09	8.73E–06	4.33E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	6.47E–14	^{123}Te	5.12E–17	8.38E–05	4.29E–21	2.14E–17	6.47E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	9.54E–02	^{125m}Te	1.43E–05	8.38E–05	1.20E–09	5.99E–06	9.54E–02	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	1.49E–06	^{227}Th	1.18E–09	8.38E–05	9.90E–14	4.94E–10	1.49E–06	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	9.05E–05	^{228}Th	7.16E–08	8.38E–05	6.00E–12	2.99E–08	9.05E–05	0.00%

Table A-8. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{229}Th	5.61E–12	1.8	1.01E–11	7.08E–09	^{229}Th	5.61E–12	8.38E–05	4.70E–16	2.34E–12	7.09E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	2.97E–05	^{230}Th	2.35E–08	8.38E–05	1.97E–12	9.84E–09	2.97E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	1.00E–03	^{231}Th	7.93E–07	8.38E–05	6.64E–11	3.31E–07	1.00E–03	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	2.34E–11	^{232}Th	1.85E–14	8.38E–05	1.55E–18	7.73E–15	2.34E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	1.03E–03	^{234}Th	8.17E–07	8.38E–05	6.84E–11	3.41E–07	1.03E–03	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	1.51E–06	^{207}Tl	1.19E–09	8.38E–05	1.00E–13	4.99E–10	1.51E–06	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	3.26E–05	^{208}Tl	2.58E–08	8.38E–05	2.16E–12	1.08E–08	3.26E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	1.53E–10	^{209}Tl	1.21E–13	8.38E–05	1.02E–17	5.06E–14	1.53E–10	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	6.06E–13	^{171}Tm	4.80E–16	8.38E–05	4.02E–20	2.01E–16	6.06E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	8.73E–05	^{232}U	6.91E–08	8.38E–05	5.79E–12	2.89E–08	8.73E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	2.24E–06	^{233}U	1.77E–09	8.38E–05	1.49E–13	7.42E–10	2.24E–06	0.00%
^{234}U	[d]		2.98E–06	2.09E–03	^{234}U	[e]		1.64E–09	8.18E–06	2.10E–03	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	1.00E–03	^{235}U	[e]		1.06E–10	5.29E–07	1.00E–03	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	2.34E–03	^{236}U	1.85E–06	8.38E–05	1.55E–10	7.73E–07	2.34E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	2.71E–04	^{237}U	2.15E–07	8.38E–05	1.80E–11	8.98E–08	2.71E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	1.03E–03	^{238}U	[e]		7.40E–11	3.69E–07	1.03E–03	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	3.71E–11	^{240}U	2.94E–14	8.38E–05	2.46E–18	1.23E–14	3.71E–11	0.00%
^{90}Y	[d]		1.87E–02	1.31E+01	^{90}Y	8.88E–01	8.38E–05	7.44E–05	3.71E–01	1.35E+01	1.22%
^{93}Zr	1.06E–04	1.8	1.90E–04	1.34E–01	^{93}Zr	1.06E–04	8.38E–05	8.86E–09	4.42E–05	1.34E–01	0.01%
			Total	1,355				Total	8	1,363	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-183.

Table A-9. Post-decontamination estimated inventory for Tank WM-184.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	5.63E–09	^{225}Ac	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	1.20E–06	^{227}Ac	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	1.73E–11	^{228}Ac	1.73E–14	2.04E–05	3.52E–19	1.76E–15	1.73E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	2.86E–10	^{108}Ag	2.85E–13	2.04E–05	5.81E–18	2.90E–14	2.86E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	3.21E–09	^{108m}Ag	[e]		2.06E–09	1.03E–05	1.03E–05	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	2.39E–14	^{109m}Ag	2.38E–17	2.04E–05	4.85E–22	2.42E–18	2.39E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	8.96E–16	^{110}Ag	8.93E–19	2.04E–05	1.82E–23	9.09E–20	8.96E–16	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	6.74E–14	^{110m}Ag	6.71E–17	2.04E–05	1.37E–21	6.83E–18	6.74E–14	0.00%
^{241}Am	[d]		3.40E–04	1.90E–01	^{241}Am	[e]		2.12E–08	1.06E–04	1.90E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	7.16E–04	^{242}Am	7.13E–07	2.04E–05	1.46E–11	7.26E–08	7.16E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	7.20E–04	^{242m}Am	7.17E–07	2.04E–05	1.46E–11	7.30E–08	7.20E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	9.87E–04	^{243}Am	9.83E–07	2.04E–05	2.01E–11	1.00E–07	9.87E–04	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	5.63E–09	^{217}At	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{137m}Ba	[d]		9.20E–01	5.13E+02	^{137m}Ba	[e]		2.04E–05	1.02E–01	5.13E+02	47.63%
^{10}Be	7.56E–11	1.8	1.36E–10	7.59E–08	^{10}Be	7.56E–11	2.04E–05	1.54E–15	7.70E–12	7.59E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	1.12E–07	^{210}Bi	1.11E–10	2.04E–05	2.27E–15	1.13E–11	1.12E–07	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	5.43E–21	^{210m}Bi	5.41E–24	2.04E–05	1.10E–28	5.51E–25	5.43E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	1.20E–06	^{211}Bi	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	7.21E–05	^{212}Bi	7.18E–08	2.04E–05	1.47E–12	7.31E–09	7.21E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	5.63E–09	^{213}Bi	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	2.89E–07	^{214}Bi	2.88E–10	2.04E–05	5.87E–15	2.93E–11	2.89E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	2.21E–06	^{14}C	2.2E–09	2.04E–05	4.49E–14	2.24E–10	2.21E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	2.39E–14	^{109}Cd	2.38E–17	2.04E–05	4.85E–22	2.42E–18	2.39E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	5.80E–02	^{113m}Cd	5.78E–05	2.04E–05	1.18E–09	5.88E–06	5.80E–02	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	7.34E–07	^{142}Ce	7.31E–10	2.04E–05	1.49E–14	7.44E–11	7.34E–07	0.00%

Table A-9. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{144}Ce	2.33E–10	1.8	4.19E–10	2.34E–07	^{144}Ce	2.33E–10	2.04E–05	4.75E–15	2.37E–11	2.34E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	7.23E–13	^{249}Cf	7.21E–16	2.04E–05	1.47E–20	7.33E–17	7.23E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	3.74E–13	^{250}Cf	3.73E–16	2.04E–05	7.61E–21	3.80E–17	3.74E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	1.15E–14	^{251}Cf	1.14E–17	2.04E–05	2.33E–22	1.16E–18	1.15E–14	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	4.86E–16	^{252}Cf	4.84E–19	2.04E–05	9.88E–24	4.93E–20	4.86E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	5.93E–04	^{242}Cm	5.91E–07	2.04E–05	1.21E–11	6.02E–08	5.94E–04	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	1.29E–04	^{243}Cm	1.29E–07	2.04E–05	2.62E–12	1.31E–08	1.29E–04	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	7.06E–03	^{244}Cm	7.03E–06	2.04E–05	1.43E–10	7.16E–07	7.06E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	1.69E–06	^{245}Cm	1.68E–09	2.04E–05	3.43E–14	1.71E–10	1.69E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	1.11E–07	^{246}Cm	1.10E–10	2.04E–05	2.25E–15	1.12E–11	1.11E–07	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	1.22E–13	^{247}Cm	1.22E–16	2.04E–05	2.49E–21	1.24E–17	1.22E–13	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	1.29E–13	^{248}Cm	1.29E–16	2.04E–05	2.63E–21	1.31E–17	1.29E–13	0.00%
^{60}Co	[d]		5.02E–05	2.80E–02	^{60}Co	[d]		3.53E–09	1.76E–05	2.80E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	3.04E–02	^{134}Cs	3.03E–05	2.04E–05	6.19E–10	3.09E–06	3.04E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	1.44E–02	^{135}Cs	1.44E–05	2.04E–05	2.93E–10	1.46E–06	1.44E–02	0.00%
^{137}Cs	[d]		9.20E–01	5.13E+02	^{137}Cs	[e]		2.04E–05	1.02E–01	5.13E+02	47.63%
^{150}Eu	2.96E–10	1.8	5.34E–10	2.98E–07	^{150}Eu	2.96E–10	2.04E–05	6.05E–15	3.02E–11	2.98E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	3.93E–02	^{152}Eu	3.92E–05	2.04E–05	7.99E–10	3.99E–06	3.93E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	1.75E+00	^{154}Eu	[e]		1.60E–08	7.98E–05	1.75E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	4.75E–01	^{155}Eu	[e]		1.45E–09	7.23E–06	4.75E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	4.90E–01	^{55}Fe	4.88E–04	2.04E–05	9.96E–09	4.97E–05	4.90E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	5.63E–09	^{221}Fr	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	1.66E–08	^{223}Fr	1.65E–11	2.04E–05	3.37E–16	1.68E–12	1.66E–08	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	3.59E–14	^{152}Gd	3.58E–17	2.04E–05	7.30E–22	3.64E–18	3.59E–14	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	4.17E–16	^{153}Gd	4.15E–19	2.04E–05	8.47E–24	4.23E–20	4.17E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	3.23E–01	^{3}H	[e]		3.20E–09	1.60E–05	3.23E–01	0.03%

Table A-9. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{166m}Ho	1.13E–09	1.8	2.03E–09	1.13E–06	^{166m}Ho	1.13E–09	2.04E–05	2.30E–14	1.15E–10	1.13E–06	0.00%
^{129}I	[d]		6.24E–07	3.48E–04	^{129}I	[e]		8.14E–11	4.06E–07	3.48E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	2.76E–13	^{115}In	2.75E–16	2.04E–05	5.61E–21	2.80E–17	2.76E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	4.70E–12	^{138}La	4.68E–15	2.04E–05	9.56E–20	4.77E–16	4.70E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	8.77E–02	^{93m}Nb	8.74E–05	2.04E–05	1.78E–09	8.89E–06	8.77E–02	0.01%
^{94}Nb	[d]		1.66E–04	9.26E–02	^{94}Nb	[e]		2.35E–08	1.17E–04	9.27E–02	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	3.98E–11	^{144}Nd	3.96E–14	2.04E–05	8.09E–19	4.03E–15	3.98E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	1.13E–02	^{59}Ni	1.13E–05	2.04E–05	2.30E–10	1.15E–06	1.13E–02	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	1.29E+00	^{63}Ni	[e]		2.67E–09	1.33E–05	1.29E+00	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	2.21E–07	^{236}Np	2.21E–10	2.04E–05	4.50E–15	2.24E–11	2.21E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	2.12E–02	^{237}Np	[e]		1.74E–09	8.68E–06	2.12E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	3.60E–06	^{238}Np	3.58E–09	2.04E–05	7.31E–14	3.65E–10	3.60E–06	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	9.87E–04	^{239}Np	9.83E–07	2.04E–05	2.01E–11	1.00E–07	9.87E–04	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	2.95E–11	^{240m}Np	2.94E–14	2.04E–05	5.99E–19	2.99E–15	2.95E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	2.13E–06	^{231}Pa	2.12E–09	2.04E–05	4.33E–14	2.16E–10	2.13E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	2.12E–02	^{233}Pa	2.11E–05	2.04E–05	4.30E–10	2.15E–06	2.12E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	1.07E–06	^{234}Pa	1.06E–09	2.04E–05	2.17E–14	1.08E–10	1.07E–06	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	8.20E–04	^{234m}Pa	8.17E–07	2.04E–05	1.67E–11	8.31E–08	8.20E–04	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	5.63E–09	^{209}Pb	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	1.12E–07	^{210}Pb	1.11E–10	2.04E–05	2.27E–15	1.13E–11	1.12E–07	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	1.20E–06	^{211}Pb	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	7.21E–05	^{212}Pb	7.18E–08	2.04E–05	1.47E–12	7.31E–09	7.21E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	2.89E–07	^{214}Pb	2.88E–10	2.04E–05	5.87E–15	2.93E–11	2.89E–07	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	4.07E–04	^{107}Pd	4.06E–07	2.04E–05	8.28E–12	4.13E–08	4.07E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	4.01E–04	^{146}Pm	4.00E–07	2.04E–05	8.16E–12	4.07E–08	4.01E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	3.90E–01	^{147}Pm	3.88E–04	2.04E–05	7.92E–09	3.95E–05	3.90E–01	0.04%

Table A-9. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{210}Po	1.08E–10	1.8	1.94E–10	1.08E–07	^{210}Po	1.08E–10	2.04E–05	2.20E–15	1.10E–11	1.08E–07	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	3.37E–09	^{211}Po	3.35E–12	2.04E–05	6.84E–17	3.41E–13	3.37E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	4.62E–05	^{212}Po	4.60E–08	2.04E–05	9.39E–13	4.68E–09	4.62E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	5.51E–09	^{213}Po	5.49E–12	2.04E–05	1.12E–16	5.59E–13	5.51E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	2.89E–07	^{214}Po	2.87E–10	2.04E–05	5.86E–15	2.93E–11	2.89E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	1.20E–06	^{215}Po	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	7.21E–05	^{216}Po	7.18E–08	2.04E–05	1.47E–12	7.31E–09	7.21E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	2.89E–07	^{218}Po	2.88E–10	2.04E–05	5.87E–15	2.93E–11	2.89E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	2.34E–07	^{144}Pr	2.33E–10	2.04E–05	4.75E–15	2.37E–11	2.34E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	2.80E–09	^{144m}Pr	2.79E–12	2.04E–05	5.70E–17	2.84E–13	2.80E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	6.55E–06	^{236}Pu	6.52E–09	2.04E–05	1.33E–13	6.64E–10	6.55E–06	0.00%
^{238}Pu	[d]		9.23E–03	5.15E+00	^{238}Pu	[e]		2.78E–06	1.39E–02	5.16E+00	0.48%
^{239}Pu	[d]		2.75E–03	1.53E+00	^{239}Pu	[e]		6.06E–07	3.02E–03	1.54E+00	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	6.08E–01	^{240}Pu	6.06E–04	2.04E–05	1.24E–08	6.17E–05	6.08E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	8.78E+00	^{241}Pu	[e]		3.80E–06	1.90E–02	8.80E+00	0.82%
^{242}Pu	4.43E–07	1.8	7.98E–07	4.45E–04	^{242}Pu	4.43E–07	2.04E–05	9.04E–12	4.51E–08	4.45E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	1.22E–13	^{243}Pu	1.22E–16	2.04E–05	2.49E–21	1.24E–17	1.22E–13	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	2.95E–11	^{244}Pu	2.94E–14	2.04E–05	6.00E–19	2.99E–15	2.95E–11	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	1.20E–06	^{223}Ra	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	7.21E–05	^{224}Ra	7.18E–08	2.04E–05	1.47E–12	7.31E–09	7.21E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	5.63E–09	^{225}Ra	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	2.89E–07	^{226}Ra	2.88E–10	2.04E–05	5.87E–15	2.93E–11	2.89E–07	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	1.73E–11	^{228}Ra	1.73E–14	2.04E–05	3.52E–19	1.76E–15	1.73E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	7.17E–07	^{87}Rb	7.15E–10	2.04E–05	1.46E–14	7.27E–11	7.17E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	2.47E–06	^{102}Rh	2.46E–09	2.04E–05	5.03E–14	2.51E–10	2.47E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	7.71E–06	^{106}Rh	7.69E–09	2.04E–05	1.57E–13	7.82E–10	7.71E–06	0.00%

Table A-9. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{219}Rn	1.20E–09	1.8	2.16E–09	1.20E–06	^{219}Rn	1.20E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	7.21E–05	^{220}Rn	7.18E–08	2.04E–05	1.47E–12	7.31E–09	7.21E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	2.89E–07	^{222}Rn	2.88E–10	2.04E–05	5.87E–15	2.93E–11	2.89E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	7.71E–06	^{106}Ru	7.69E–09	2.04E–05	1.57E–13	7.82E–10	7.71E–06	0.00%
^{125}Sb	[d]		5.55E–04	3.09E–01	^{125}Sb	[e]		1.55E–08	7.73E–05	3.10E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	1.42E–03	^{126}Sb	1.41E–06	2.04E–05	2.89E–11	1.44E–07	1.42E–03	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	1.01E–02	^{126m}Sb	1.01E–05	2.04E–05	2.06E–10	1.03E–06	1.01E–02	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	1.08E–02	^{79}Se	1.07E–05	2.04E–05	2.19E–10	1.09E–06	1.08E–02	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	4.66E–09	^{146}Sm	4.65E–12	2.04E–05	9.48E–17	4.73E–13	4.66E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	1.82E–07	^{147}Sm	1.81E–10	2.04E–05	3.70E–15	1.85E–11	1.82E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	9.34E–13	^{148}Sm	9.30E–16	2.04E–05	1.90E–20	9.47E–17	9.34E–13	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	8.29E–14	^{149}Sm	8.26E–17	2.04E–05	1.68E–21	8.41E–18	8.29E–14	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	7.74E+00	^{151}Sm	7.71E–03	2.04E–05	1.57E–07	7.84E–04	7.74E+00	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	9.58E–12	^{119m}Sn	9.55E–15	2.04E–05	1.95E–19	9.72E–16	9.58E–12	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	5.76E–02	^{121m}Sn	5.74E–05	2.04E–05	1.17E–09	5.84E–06	5.76E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	1.01E–02	^{126}Sn	1.01E–05	2.04E–05	2.06E–10	1.03E–06	1.01E–02	0.00%
^{90}Sr	[d]		1.87E–02	1.04E+01	^{90}Sr	[e]		2.55E–06	1.27E–02	1.04E+01	0.97%
^{98}Tc	6.34E–11	1.8	1.14E–10	6.37E–08	^{98}Tc	6.34E–11	2.04E–05	1.29E–15	6.46E–12	6.37E–08	0.00%
^{99}Tc	[d]		6.17E–04	3.44E–01	^{99}Tc	[e]		1.03E–09	5.14E–06	3.44E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	5.14E–14	^{123}Te	5.12E–17	2.04E–05	1.05E–21	5.21E–18	5.14E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	7.58E–02	^{125m}Te	1.43E–05	2.04E–05	2.92E–10	1.46E–06	7.58E–02	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	1.19E–06	^{227}Th	1.18E–09	2.04E–05	2.41E–14	1.20E–10	1.19E–06	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	7.19E–05	^{228}Th	7.16E–08	2.04E–05	1.46E–12	7.29E–09	7.19E–05	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	5.63E–09	^{229}Th	5.61E–12	2.04E–05	1.14E–16	5.71E–13	5.63E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	2.36E–05	^{230}Th	2.35E–08	2.04E–05	4.80E–13	2.40E–09	2.36E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	7.96E–04	^{231}Th	7.93E–07	2.04E–05	1.62E–11	8.07E–08	7.96E–04	0.00%

Table A-9. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{232}Th	1.85E–14	1.8	3.33E–14	1.86E–11	^{232}Th	1.85E–14	2.04E–05	3.77E–19	1.88E–15	1.86E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	8.20E–04	^{234}Th	8.17E–07	2.04E–05	1.67E–11	8.31E–08	8.20E–04	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	1.20E–06	^{207}Tl	1.19E–09	2.04E–05	2.44E–14	1.22E–10	1.20E–06	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	2.59E–05	^{208}Tl	2.58E–08	2.04E–05	5.27E–13	2.63E–09	2.59E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	1.22E–10	^{209}Tl	1.21E–13	2.04E–05	2.47E–18	1.23E–14	1.22E–10	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	4.81E–13	^{171}Tm	4.80E–16	2.04E–05	9.78E–21	4.88E–17	4.81E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	6.94E–05	^{232}U	6.91E–08	2.04E–05	1.41E–12	7.03E–09	6.94E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	1.78E–06	^{233}U	1.77E–09	2.04E–05	3.62E–14	1.81E–10	1.78E–06	0.00%
^{234}U	[d]		2.98E–06	1.66E–03	^{234}U	[e]		3.34E–10	1.67E–06	1.66E–03	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	7.96E–04	^{235}U	7.93E–07	2.04E–05	1.62E–11	8.07E–08	7.96E–04	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	1.86E–03	^{236}U	1.85E–06	2.04E–05	3.77E–11	1.88E–07	1.86E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	2.15E–04	^{237}U	2.15E–07	2.04E–05	4.38E–12	2.18E–08	2.15E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	8.20E–04	^{238}U	8.17E–07	2.04E–05	1.67E–11	8.31E–08	8.20E–04	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	2.95E–11	^{240}U	2.94E–14	2.04E–05	5.99E–19	2.99E–15	2.95E–11	0.00%
^{90}Y	[d]		1.87E–02	1.04E+01	^{90}Y	8.88E–01	2.04E–05	1.81E–05	9.04E–02	1.05E+01	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	1.06E–01	^{93}Zr	1.06E–04	2.04E–05	2.16E–09	1.08E–05	1.06E–01	0.01%
			Total	1,077				Total	0.27	1,077	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-184.

Table A-10. Post-decontamination estimated inventory for Tank WM-185.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	7.27E–09	^{225}Ac	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	1.55E–06	^{227}Ac	1.20E–09	2.11E–05	2.52E–14	1.26E–10	1.55E–06	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	2.24E–11	^{228}Ac	1.73E–14	2.11E–05	3.64E–19	1.82E–15	2.24E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	3.69E–10	^{108}Ag	2.85E–13	2.11E–05	6.01E–18	3.00E–14	3.69E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	4.15E–09	^{108m}Ag	3.20E–12	2.11E–05	6.76E–17	3.37E–13	4.15E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	3.08E–14	^{109m}Ag	2.38E–17	2.11E–05	5.02E–22	2.50E–18	3.08E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	1.16E–15	^{110}Ag	8.93E–19	2.11E–05	1.88E–23	9.40E–20	1.16E–15	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	8.70E–14	^{110m}Ag	6.71E–17	2.11E–05	1.42E–21	7.07E–18	8.70E–14	0.00%
^{241}Am	[d]		3.40E–04	2.45E–01	^{241}Am	[e]		2.28E–09	1.14E–05	2.45E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	9.25E–04	^{242}Am	7.13E–07	2.11E–05	1.51E–11	7.51E–08	9.25E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	9.29E–04	^{242m}Am	7.17E–07	2.11E–05	1.51E–11	7.55E–08	9.29E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	1.27E–03	^{243}Am	9.83E–07	2.11E–05	2.07E–11	1.03E–07	1.27E–03	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	7.27E–09	^{217}At	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{137m}Ba	[d]		9.20E–01	6.62E+02	^{137m}Ba	[e]		2.11E–05	1.05E–01	6.63E+02	47.64%
^{10}Be	7.56E–11	1.8	1.36E–10	9.80E–08	^{10}Be	7.56E–11	2.11E–05	1.60E–15	7.96E–12	9.80E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	1.44E–07	^{210}Bi	1.11E–10	2.11E–05	2.35E–15	1.17E–11	1.44E–07	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	7.02E–21	^{210m}Bi	5.41E–24	2.11E–05	1.14E–28	5.70E–25	7.02E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	1.55E–06	^{211}Bi	1.20E–09	2.11E–05	2.53E–14	1.26E–10	1.55E–06	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	9.31E–05	^{212}Bi	7.18E–08	2.11E–05	1.52E–12	7.56E–09	9.31E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	7.27E–09	^{213}Bi	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	3.73E–07	^{214}Bi	2.88E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	2.85E–06	^{14}C	2.2E–09	2.11E–05	4.64E–14	2.32E–10	2.85E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	3.08E–14	^{109}Cd	2.38E–17	2.11E–05	5.02E–22	2.50E–18	3.08E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	7.49E–02	^{113m}Cd	5.78E–05	2.11E–05	1.22E–09	6.08E–06	7.49E–02	0.01%
^{142}Ce	7.31E–10	1.8	1.32E–09	9.48E–07	^{142}Ce	7.31E–10	2.11E–05	1.54E–14	7.70E–11	9.48E–07	0.00%

Table A-10. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{144}Ce	2.33E–10	1.8	4.19E–10	3.02E–07	^{144}Ce	2.33E–10	2.11E–05	4.91E–15	2.45E–11	3.02E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	9.34E–13	^{249}Cf	7.21E–16	2.11E–05	1.52E–20	7.58E–17	9.34E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	4.83E–13	^{250}Cf	3.73E–16	2.11E–05	7.87E–21	3.93E–17	4.83E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	1.48E–14	^{251}Cf	1.14E–17	2.11E–05	2.41E–22	1.20E–18	1.48E–14	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	6.28E–16	^{252}Cf	4.84E–19	2.11E–05	1.02E–23	5.10E–20	6.28E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	7.66E–04	^{242}Cm	5.91E–07	2.11E–05	1.25E–11	6.22E–08	7.66E–04	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	1.67E–04	^{243}Cm	1.29E–07	2.11E–05	2.71E–12	1.35E–08	1.67E–04	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	9.11E–03	^{244}Cm	7.03E–06	2.11E–05	1.48E–10	7.40E–07	9.12E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	2.18E–06	^{245}Cm	1.68E–09	2.11E–05	3.55E–14	1.77E–10	2.18E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	1.43E–07	^{246}Cm	1.10E–10	2.11E–05	2.32E–15	1.16E–11	1.43E–07	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	1.58E–13	^{247}Cm	1.22E–16	2.11E–05	2.57E–21	1.28E–17	1.58E–13	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	1.67E–13	^{248}Cm	1.29E–16	2.11E–05	2.72E–21	1.36E–17	1.67E–13	0.00%
^{60}Co	[d]		5.02E–05	3.61E–02	^{60}Co	3.70E–04	2.11E–05	7.82E–09	3.90E–05	3.62E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	3.93E–02	^{134}Cs	3.03E–05	2.11E–05	6.40E–10	3.19E–06	3.93E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	1.86E–02	^{135}Cs	1.44E–05	2.11E–05	3.03E–10	1.51E–06	1.86E–02	0.00%
^{137}Cs	[d]		9.20E–01	6.62E+02	^{137}Cs	[e]		2.11E–05	1.05E–01	6.63E+02	47.64%
^{150}Eu	2.96E–10	1.8	5.34E–10	3.84E–07	^{150}Eu	2.96E–10	2.11E–05	6.26E–15	3.12E–11	3.84E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	5.08E–02	^{152}Eu	3.92E–05	2.11E–05	8.27E–10	4.12E–06	5.08E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	2.26E+00	^{154}Eu	[e]		1.94E–08	9.68E–05	2.26E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	6.14E–01	^{155}Eu	4.74E–04	2.11E–05	9.99E–09	4.99E–05	6.14E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	6.33E–01	^{55}Fe	4.88E–04	2.11E–05	1.03E–08	5.14E–05	6.33E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	7.27E–09	^{221}Fr	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	2.14E–08	^{223}Fr	1.65E–11	2.11E–05	3.48E–16	1.74E–12	2.14E–08	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	4.64E–14	^{152}Gd	3.58E–17	2.11E–05	7.55E–22	3.77E–18	4.64E–14	0.00%
^{153}Gd	4.15E–19	1.8	7.48E–19	5.38E–16	^{153}Gd	4.15E–19	2.11E–05	8.76E–24	4.37E–20	5.38E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	4.17E–01	^{3}H	[e]		4.01E–09	2.00E–05	4.17E–01	0.03%

Table A-10. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{166m}Ho	1.13E–09	1.8	2.03E–09	1.46E–06	^{166m}Ho	1.13E–09	2.11E–05	2.38E–14	1.19E–10	1.46E–06	0.00%
^{129}I	[d]		6.24E–07	4.49E–04	^{129}I	[e]		1.94E–10	9.68E–07	4.50E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	3.56E–13	^{115}In	2.75E–16	2.11E–05	5.80E–21	2.90E–17	3.56E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	6.07E–12	^{138}La	4.68E–15	2.11E–05	9.88E–20	4.93E–16	6.07E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	1.13E–01	^{93m}Nb	8.74E–05	2.11E–05	1.84E–09	9.20E–06	1.13E–01	0.01%
^{94}Nb	[d]		1.66E–04	1.20E–01	^{94}Nb	3.62E–05	2.11E–05	7.64E–10	3.81E–06	1.20E–01	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	5.14E–11	^{144}Nd	3.96E–14	2.11E–05	8.36E–19	4.17E–15	5.14E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	1.46E–02	^{59}Ni	1.13E–05	2.11E–05	2.38E–10	1.19E–06	1.46E–02	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	1.66E+00	^{63}Ni	[e]		7.15E–09	3.57E–05	1.66E+00	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	2.86E–07	^{236}Np	2.21E–10	2.11E–05	4.65E–15	2.32E–11	2.86E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	2.73E–02	^{237}Np	[e]		9.61E–11	4.79E–07	2.73E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	4.65E–06	^{238}Np	3.58E–09	2.11E–05	7.56E–14	3.77E–10	4.65E–06	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	1.27E–03	^{239}Np	9.83E–07	2.11E–05	2.07E–11	1.03E–07	1.27E–03	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	3.81E–11	^{240m}Np	2.94E–14	2.11E–05	6.20E–19	3.09E–15	3.81E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	2.75E–06	^{231}Pa	2.12E–09	2.11E–05	4.47E–14	2.23E–10	2.75E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	2.73E–02	^{233}Pa	2.11E–05	2.11E–05	4.45E–10	2.22E–06	2.74E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	1.38E–06	^{234}Pa	1.06E–09	2.11E–05	2.24E–14	1.12E–10	1.38E–06	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	1.06E–03	^{234m}Pa	8.17E–07	2.11E–05	1.72E–11	8.60E–08	1.06E–03	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	7.27E–09	^{209}Pb	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	1.44E–07	^{210}Pb	1.11E–10	2.11E–05	2.35E–15	1.17E–11	1.44E–07	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	1.55E–06	^{211}Pb	1.20E–09	2.11E–05	2.53E–14	1.26E–10	1.55E–06	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	9.31E–05	^{212}Pb	7.18E–08	2.11E–05	1.52E–12	7.56E–09	9.31E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	3.73E–07	^{214}Pb	2.88E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{107}Pd	4.06E–07	1.8	7.31E–07	5.26E–04	^{107}Pd	4.06E–07	2.11E–05	8.56E–12	4.27E–08	5.26E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	5.18E–04	^{146}Pm	4.00E–07	2.11E–05	8.44E–12	4.21E–08	5.18E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	5.03E–01	^{147}Pm	3.88E–04	2.11E–05	8.19E–09	4.09E–05	5.03E–01	0.04%

Table A-10. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{210}Po	1.08E–10	1.8	1.94E–10	1.39E–07	^{210}Po	1.08E–10	2.11E–05	2.27E–15	1.13E–11	1.39E–07	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	4.35E–09	^{211}Po	3.35E–12	2.11E–05	7.08E–17	3.53E–13	4.35E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	5.97E–05	^{212}Po	4.60E–08	2.11E–05	9.71E–13	4.85E–09	5.97E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	7.11E–09	^{213}Po	5.49E–12	2.11E–05	1.16E–16	5.78E–13	7.11E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	3.73E–07	^{214}Po	2.87E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	1.55E–06	^{215}Po	1.20E–09	2.11E–05	2.53E–14	1.26E–10	1.55E–06	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	9.31E–05	^{216}Po	7.18E–08	2.11E–05	1.52E–12	7.56E–09	9.31E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	3.73E–07	^{218}Po	2.88E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	3.02E–07	^{144}Pr	2.33E–10	2.11E–05	4.91E–15	2.45E–11	3.02E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	3.62E–09	^{144m}Pr	2.79E–12	2.11E–05	5.89E–17	2.94E–13	3.62E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	8.45E–06	^{236}Pu	6.52E–09	2.11E–05	1.38E–13	6.87E–10	8.46E–06	0.00%
^{238}Pu	[d]		9.23E–03	6.65E+00	^{238}Pu	[e]		8.59E–08	4.29E–04	6.65E+00	0.48%
^{239}Pu	[d]		2.75E–03	1.98E+00	^{239}Pu	[e]		6.82E–09	3.40E–05	1.98E+00	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	7.85E–01	^{240}Pu	6.06E–04	2.11E–05	1.28E–08	6.38E–05	7.85E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	1.13E+01	^{241}Pu	[e]		8.95E–08	4.47E–04	1.13E+01	0.82%
^{242}Pu	4.43E–07	1.8	7.98E–07	5.74E–04	^{242}Pu	4.43E–07	2.11E–05	9.35E–12	4.66E–08	5.74E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	1.58E–13	^{243}Pu	1.22E–16	2.11E–05	2.57E–21	1.28E–17	1.58E–13	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	3.81E–11	^{244}Pu	2.94E–14	2.11E–05	6.20E–19	3.09E–15	3.81E–11	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	1.55E–06	^{223}Ra	1.20E–09	2.11E–05	2.53E–14	1.26E–10	1.55E–06	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	9.31E–05	^{224}Ra	7.18E–08	2.11E–05	1.52E–12	7.56E–09	9.31E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	7.27E–09	^{225}Ra	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	3.73E–07	^{226}Ra	2.88E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{228}Ra	1.73E–14	1.8	3.11E–14	2.24E–11	^{228}Ra	1.73E–14	2.11E–05	3.64E–19	1.82E–15	2.24E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	9.26E–07	^{87}Rb	7.15E–10	2.11E–05	1.51E–14	7.52E–11	9.26E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	3.19E–06	^{102}Rh	2.46E–09	2.11E–05	5.20E–14	2.59E–10	3.19E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	9.96E–06	^{106}Rh	7.69E–09	2.11E–05	1.62E–13	8.09E–10	9.96E–06	0.00%

Table A-10. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{219}Rn	1.20E–09	1.8	2.16E–09	1.55E–06	^{219}Rn	1.20E–09	2.11E–05	2.53E–14	1.26E–10	1.55E–06	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	9.31E–05	^{220}Rn	7.18E–08	2.11E–05	1.52E–12	7.56E–09	9.31E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	3.73E–07	^{222}Rn	2.88E–10	2.11E–05	6.07E–15	3.03E–11	3.73E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	9.96E–06	^{106}Ru	7.69E–09	2.11E–05	1.62E–13	8.09E–10	9.96E–06	0.00%
^{125}Sb	[d]		5.55E–04	4.00E–01	^{125}Sb	5.87E–05	2.11E–05	1.24E–09	6.18E–06	4.00E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	1.83E–03	^{126}Sb	1.41E–06	2.11E–05	2.98E–11	1.49E–07	1.83E–03	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	1.31E–02	^{126m}Sb	1.01E–05	2.11E–05	2.13E–10	1.06E–06	1.31E–02	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	1.39E–02	^{79}Se	1.07E–05	2.11E–05	2.26E–10	1.13E–06	1.39E–02	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	6.02E–09	^{146}Sm	4.65E–12	2.11E–05	9.80E–17	4.89E–13	6.02E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	2.35E–07	^{147}Sm	1.81E–10	2.11E–05	3.83E–15	1.91E–11	2.35E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	1.21E–12	^{148}Sm	9.30E–16	2.11E–05	1.96E–20	9.79E–17	1.21E–12	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	1.07E–13	^{149}Sm	8.26E–17	2.11E–05	1.74E–21	8.69E–18	1.07E–13	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	9.99E+00	^{151}Sm	7.71E–03	2.11E–05	1.63E–07	8.11E–04	9.99E+00	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	1.24E–11	^{119m}Sn	9.55E–15	2.11E–05	2.01E–19	1.00E–15	1.24E–11	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	7.44E–02	^{121m}Sn	5.74E–05	2.11E–05	1.21E–09	6.04E–06	7.44E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	1.31E–02	^{126}Sn	1.01E–05	2.11E–05	2.13E–10	1.06E–06	1.31E–02	0.00%
^{90}Sr	[d]		1.87E–02	1.35E+01	^{90}Sr	[e]		1.98E–06	9.88E–03	1.35E+01	0.97%
^{98}Tc	6.34E–11	1.8	1.14E–10	8.22E–08	^{98}Tc	6.34E–11	2.11E–05	1.34E–15	6.68E–12	8.22E–08	0.00%
^{99}Tc	[d]		6.17E–04	4.44E–01	^{99}Tc	[e]		3.69E–10	1.84E–06	4.44E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	6.64E–14	^{123}Te	5.12E–17	2.11E–05	1.08E–21	5.39E–18	6.64E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	9.79E–02	^{125m}Te	1.43E–05	2.11E–05	3.02E–10	1.51E–06	9.79E–02	0.01%
^{227}Th	1.18E–09	1.8	2.13E–09	1.53E–06	^{227}Th	1.18E–09	2.11E–05	2.49E–14	1.24E–10	1.53E–06	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	9.28E–05	^{228}Th	7.16E–08	2.11E–05	1.51E–12	7.54E–09	9.28E–05	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	7.27E–09	^{229}Th	5.61E–12	2.11E–05	1.18E–16	5.90E–13	7.27E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	3.05E–05	^{230}Th	2.35E–08	2.11E–05	4.97E–13	2.48E–09	3.05E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	1.03E–03	^{231}Th	7.93E–07	2.11E–05	1.67E–11	8.35E–08	1.03E–03	0.00%

Table A-10. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{232}Th	1.85E–14	1.8	3.33E–14	2.40E–11	^{232}Th	1.85E–14	2.11E–05	3.90E–19	1.95E–15	2.40E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	1.06E–03	^{234}Th	8.17E–07	2.11E–05	1.72E–11	8.60E–08	1.06E–03	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	1.55E–06	^{207}Tl	1.19E–09	2.11E–05	2.52E–14	1.26E–10	1.55E–06	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	3.35E–05	^{208}Tl	2.58E–08	2.11E–05	5.45E–13	2.72E–09	3.35E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	1.57E–10	^{209}Tl	1.21E–13	2.11E–05	2.56E–18	1.28E–14	1.57E–10	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	6.22E–13	^{171}Tm	4.80E–16	2.11E–05	1.01E–20	5.05E–17	6.22E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	8.96E–05	^{232}U	6.91E–08	2.11E–05	1.46E–12	7.27E–09	8.96E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	2.30E–06	^{233}U	1.77E–09	2.11E–05	3.74E–14	1.87E–10	2.30E–06	0.00%
^{234}U	[d]		2.98E–06	2.15E–03	^{234}U	3.37E–05	2.11E–05	7.11E–10	3.55E–06	2.15E–03	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	1.03E–03	^{235}U	7.93E–07	2.11E–05	1.67E–11	8.35E–08	1.03E–03	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	2.40E–03	^{236}U	1.85E–06	2.11E–05	3.90E–11	1.95E–07	2.40E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	2.78E–04	^{237}U	2.15E–07	2.11E–05	4.53E–12	2.26E–08	2.78E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	1.06E–03	^{238}U	8.17E–07	2.11E–05	1.72E–11	8.60E–08	1.06E–03	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	3.81E–11	^{240}U	2.94E–14	2.11E–05	6.20E–19	3.09E–15	3.81E–11	0.00%
^{90}Y	[d]		1.87E–02	1.35E+01	^{90}Y	8.88E–01	2.11E–05	1.87E–05	9.35E–02	1.36E+01	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	1.37E–01	^{93}Zr	1.06E–04	2.11E–05	2.23E–09	1.11E–05	1.37E–01	0.01%
			Total	1,391				Total	0.23	1,391	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-185.

Table A-11. Post-decontamination estimated inventory for Tank WM-186.

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{225}Ac	5.61E–12	1.8	1.01E–11	3.37E–09	^{225}Ac	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{227}Ac	1.20E–09	1.8	2.15E–09	7.19E–07	^{227}Ac	1.20E–09	2.88E–06	3.44E–15	1.72E–11	7.19E–07	0.00%
^{228}Ac	1.73E–14	1.8	3.11E–14	1.04E–11	^{228}Ac	1.73E–14	2.88E–06	4.97E–20	2.48E–16	1.04E–11	0.00%
^{108}Ag	2.85E–13	1.8	5.13E–13	1.71E–10	^{108}Ag	2.85E–13	2.88E–06	8.21E–19	4.09E–15	1.71E–10	0.00%
^{108m}Ag	3.20E–12	1.8	5.76E–12	1.93E–09	^{108m}Ag	3.20E–12	2.88E–06	9.22E–18	4.60E–14	1.93E–09	0.00%
^{109m}Ag	2.38E–17	1.8	4.28E–17	1.43E–14	^{109m}Ag	2.38E–17	2.88E–06	6.85E–23	3.42E–19	1.43E–14	0.00%
^{110}Ag	8.93E–19	1.8	1.61E–18	5.37E–16	^{110}Ag	8.93E–19	2.88E–06	2.57E–24	1.28E–20	5.37E–16	0.00%
^{110m}Ag	6.71E–17	1.8	1.21E–16	4.04E–14	^{110m}Ag	6.71E–17	2.88E–06	1.93E–22	9.65E–19	4.04E–14	0.00%
^{241}Am	[d]		3.40E–04	1.14E–01	^{241}Am	[e]		1.14E–09	5.69E–06	1.14E–01	0.02%
^{242}Am	7.13E–07	1.8	1.28E–06	4.29E–04	^{242}Am	7.13E–07	2.88E–06	2.05E–12	1.03E–08	4.29E–04	0.00%
^{242m}Am	7.17E–07	1.8	1.29E–06	4.32E–04	^{242m}Am	7.17E–07	2.88E–06	2.06E–12	1.03E–08	4.32E–04	0.00%
^{243}Am	9.83E–07	1.8	1.77E–06	5.92E–04	^{243}Am	9.83E–07	2.88E–06	2.83E–12	1.41E–08	5.92E–04	0.00%
^{217}At	5.61E–12	1.8	1.01E–11	3.37E–09	^{217}At	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{137m}Ba	[b]		9.20E–01	3.08E+02	^{137m}Ba	[e]		2.88E–06	1.44E–02	3.08E+02	47.63%
^{10}Be	7.56E–11	1.8	1.36E–10	4.55E–08	^{10}Be	7.56E–11	2.88E–06	2.18E–16	1.09E–12	4.55E–08	0.00%
^{210}Bi	1.11E–10	1.8	2.01E–10	6.71E–08	^{210}Bi	1.11E–10	2.88E–06	3.21E–16	1.60E–12	6.71E–08	0.00%
^{210m}Bi	5.41E–24	1.8	9.75E–24	3.26E–21	^{210m}Bi	5.41E–24	2.88E–06	1.56E–29	7.78E–26	3.26E–21	0.00%
^{211}Bi	1.20E–09	1.8	2.16E–09	7.21E–07	^{211}Bi	1.20E–09	2.88E–06	3.45E–15	1.72E–11	7.21E–07	0.00%
^{212}Bi	7.18E–08	1.8	1.29E–07	4.32E–05	^{212}Bi	7.18E–08	2.88E–06	2.07E–13	1.03E–09	4.32E–05	0.00%
^{213}Bi	5.61E–12	1.8	1.01E–11	3.37E–09	^{213}Bi	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{214}Bi	2.88E–10	1.8	5.18E–10	1.73E–07	^{214}Bi	2.88E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%
^{14}C	2.20E–09	1.8	3.96E–09	1.32E–06	^{14}C	2.20E–09	2.88E–06	6.34E–15	3.16E–11	1.32E–06	0.00%
^{109}Cd	2.38E–17	1.8	4.28E–17	1.43E–14	^{109}Cd	2.38E–17	2.88E–06	6.85E–23	3.42E–19	1.43E–14	0.00%
^{113m}Cd	5.78E–05	1.8	1.04E–04	3.48E–02	^{113m}Cd	5.78E–05	2.88E–06	1.66E–10	8.30E–07	3.48E–02	0.01%

Table A-11. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{142}Ce	7.31E–10	1.8	1.32E–09	4.40E–07	^{142}Ce	7.31E–10	2.88E–06	2.11E–15	1.05E–11	4.40E–07	0.00%
^{144}Ce	2.33E–10	1.8	4.19E–10	1.40E–07	^{144}Ce	2.33E–10	2.88E–06	6.70E–16	3.34E–12	1.40E–07	0.00%
^{249}Cf	7.21E–16	1.8	1.30E–15	4.34E–13	^{249}Cf	7.21E–16	2.88E–06	2.08E–21	1.04E–17	4.34E–13	0.00%
^{250}Cf	3.73E–16	1.8	6.71E–16	2.24E–13	^{250}Cf	3.73E–16	2.88E–06	1.07E–21	5.36E–18	2.24E–13	0.00%
^{251}Cf	1.14E–17	1.8	2.06E–17	6.88E–15	^{251}Cf	1.14E–17	2.88E–06	3.29E–23	1.64E–19	6.88E–15	0.00%
^{252}Cf	4.84E–19	1.8	8.72E–19	2.91E–16	^{252}Cf	4.84E–19	2.88E–06	1.39E–24	6.96E–21	2.91E–16	0.00%
^{242}Cm	5.91E–07	1.8	1.06E–06	3.56E–04	^{242}Cm	5.91E–07	2.88E–06	1.70E–12	8.50E–09	3.56E–04	0.00%
^{243}Cm	1.29E–07	1.8	2.31E–07	7.74E–05	^{243}Cm	1.29E–07	2.88E–06	3.70E–13	1.85E–09	7.74E–05	0.00%
^{244}Cm	7.03E–06	1.8	1.27E–05	4.23E–03	^{244}Cm	7.03E–06	2.88E–06	2.03E–11	1.01E–07	4.23E–03	0.00%
^{245}Cm	1.68E–09	1.8	3.02E–09	1.01E–06	^{245}Cm	1.68E–09	2.88E–06	4.84E–15	2.41E–11	1.01E–06	0.00%
^{246}Cm	1.10E–10	1.8	1.98E–10	6.63E–08	^{246}Cm	1.10E–10	2.88E–06	3.17E–16	1.58E–12	6.63E–08	0.00%
^{247}Cm	1.22E–16	1.8	2.20E–16	7.34E–14	^{247}Cm	1.22E–16	2.88E–06	3.51E–22	1.75E–18	7.34E–14	0.00%
^{248}Cm	1.29E–16	1.8	2.32E–16	7.75E–14	^{248}Cm	1.29E–16	2.88E–06	3.71E–22	1.85E–18	7.75E–14	0.00%
^{60}Co	[d]		5.02E–05	1.68E–02	^{60}Co	[e]		1.44E–09	7.18E–06	1.68E–02	0.00%
^{134}Cs	3.03E–05	1.8	5.46E–05	1.83E–02	^{134}Cs	3.03E–05	2.88E–06	8.74E–11	4.36E–07	1.83E–02	0.00%
^{135}Cs	1.44E–05	1.8	2.59E–05	8.66E–03	^{135}Cs	1.44E–05	2.88E–06	4.14E–11	2.07E–07	8.66E–03	0.00%
^{137}Cs	[d]		9.20E–01	3.08E+02	^{137}Cs	[e]		2.88E–06	1.44E–02	3.08E+02	47.63%
^{150}Eu	2.96E–10	1.8	5.34E–10	1.78E–07	^{150}Eu	2.96E–10	2.88E–06	8.54E–16	4.26E–12	1.78E–07	0.00%
^{152}Eu	3.92E–05	1.8	7.05E–05	2.36E–02	^{152}Eu	3.92E–05	2.88E–06	1.13E–10	5.63E–07	2.36E–02	0.00%
^{154}Eu	1.75E–03	1.8	3.15E–03	1.05E+00	^{154}Eu	[e]		6.02E–09	3.00E–05	1.05E+00	0.16%
^{155}Eu	4.74E–04	1.8	8.53E–04	2.85E–01	^{155}Eu	4.74E–04	2.88E–06	1.36E–09	6.81E–06	2.85E–01	0.04%
^{55}Fe	4.88E–04	1.8	8.79E–04	2.94E–01	^{55}Fe	4.88E–04	2.88E–06	1.41E–09	7.01E–06	2.94E–01	0.05%
^{221}Fr	5.61E–12	1.8	1.01E–11	3.37E–09	^{221}Fr	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{223}Fr	1.65E–11	1.8	2.97E–11	9.93E–09	^{223}Fr	1.65E–11	2.88E–06	4.75E–17	2.37E–13	9.93E–09	0.00%
^{152}Gd	3.58E–17	1.8	6.44E–17	2.15E–14	^{152}Gd	3.58E–17	2.88E–06	1.03E–22	5.14E–19	2.15E–14	0.00%

Table A-11. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{153}Gd	4.15E–19	1.8	7.48E–19	2.50E–16	^{153}Gd	4.15E–19	2.88E–06	1.20E–24	5.97E–21	2.50E–16	0.00%
^{3}H	3.22E–04	1.8	5.79E–04	1.94E–01	^{3}H	[e]		1.46E–09	7.28E–06	1.94E–01	0.03%
^{166m}Ho	1.13E–09	1.8	2.03E–09	6.78E–07	^{166m}Ho	1.13E–09	2.88E–06	3.24E–15	1.62E–11	6.78E–07	0.00%
^{129}I	[d]		6.24E–07	2.09E–04	^{129}I	[e]		2.89E–11	1.44E–07	2.09E–04	0.00%
^{115}In	2.75E–16	1.8	4.95E–16	1.66E–13	^{115}In	2.75E–16	2.88E–06	7.92E–22	3.95E–18	1.66E–13	0.00%
^{138}La	4.68E–15	1.8	8.43E–15	2.82E–12	^{138}La	4.68E–15	2.88E–06	1.35E–20	6.73E–17	2.82E–12	0.00%
^{93m}Nb	8.74E–05	1.8	1.57E–04	5.26E–02	^{93m}Nb	8.74E–05	2.88E–06	2.52E–10	1.26E–06	5.26E–02	0.01%
^{94}Nb	[d]		1.66E–04	5.55E–02	^{94}Nb	[e]		5.84E–09	2.91E–05	5.55E–02	0.01%
^{144}Nd	3.96E–14	1.8	7.14E–14	2.39E–11	^{144}Nd	3.96E–14	2.88E–06	1.14E–19	5.70E–16	2.39E–11	0.00%
^{59}Ni	1.13E–05	1.8	2.03E–05	6.78E–03	^{59}Ni	1.13E–05	2.88E–06	3.25E–11	1.62E–07	6.78E–03	0.00%
^{63}Ni	1.28E–03	1.8	2.31E–03	7.72E–01	^{63}Ni	[e]		5.78E–09	2.88E–05	7.72E–01	0.12%
^{236}Np	2.21E–10	1.8	3.97E–10	1.33E–07	^{236}Np	2.21E–10	2.88E–06	6.35E–16	3.17E–12	1.33E–07	0.00%
^{237}Np	2.11E–05	1.8	3.80E–05	1.27E–02	^{237}Np	[e]		6.67E–11	3.33E–07	1.27E–02	0.00%
^{238}Np	3.58E–09	1.8	6.45E–09	2.16E–06	^{238}Np	3.58E–09	2.88E–06	1.03E–14	5.15E–11	2.16E–06	0.00%
^{239}Np	9.83E–07	1.8	1.77E–06	5.92E–04	^{239}Np	9.83E–07	2.88E–06	2.83E–12	1.41E–08	5.92E–04	0.00%
^{240m}Np	2.94E–14	1.8	5.28E–14	1.77E–11	^{240m}Np	2.94E–14	2.88E–06	8.46E–20	4.22E–16	1.77E–11	0.00%
^{231}Pa	2.12E–09	1.8	3.82E–09	1.28E–06	^{231}Pa	2.12E–09	2.88E–06	6.11E–15	3.05E–11	1.28E–06	0.00%
^{233}Pa	2.11E–05	1.8	3.80E–05	1.27E–02	^{233}Pa	2.11E–05	2.88E–06	6.08E–11	3.03E–07	1.27E–02	0.00%
^{234}Pa	1.06E–09	1.8	1.91E–09	6.39E–07	^{234}Pa	1.06E–09	2.88E–06	3.06E–15	1.53E–11	6.39E–07	0.00%
^{234m}Pa	8.17E–07	1.8	1.47E–06	4.92E–04	^{234m}Pa	8.17E–07	2.88E–06	2.35E–12	1.17E–08	4.92E–04	0.00%
^{209}Pb	5.61E–12	1.8	1.01E–11	3.37E–09	^{209}Pb	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{210}Pb	1.11E–10	1.8	2.01E–10	6.71E–08	^{210}Pb	1.11E–10	2.88E–06	3.21E–16	1.60E–12	6.71E–08	0.00%
^{211}Pb	1.20E–09	1.8	2.16E–09	7.21E–07	^{211}Pb	1.20E–09	2.88E–06	3.45E–15	1.72E–11	7.21E–07	0.00%
^{212}Pb	7.18E–08	1.8	1.29E–07	4.32E–05	^{212}Pb	7.18E–08	2.88E–06	2.07E–13	1.03E–09	4.32E–05	0.00%
^{214}Pb	2.88E–10	1.8	5.18E–10	1.73E–07	^{214}Pb	2.88E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%

Table A-11. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{107}Pd	4.06E–07	1.8	7.31E–07	2.44E–04	^{107}Pd	4.06E–07	2.88E–06	1.17E–12	5.83E–09	2.44E–04	0.00%
^{146}Pm	4.00E–07	1.8	7.20E–07	2.41E–04	^{146}Pm	4.00E–07	2.88E–06	1.15E–12	5.75E–09	2.41E–04	0.00%
^{147}Pm	3.88E–04	1.8	6.99E–04	2.34E–01	^{147}Pm	3.88E–04	2.88E–06	1.12E–09	5.58E–06	2.34E–01	0.04%
^{210}Po	1.08E–10	1.8	1.94E–10	6.48E–08	^{210}Po	1.08E–10	2.88E–06	3.10E–16	1.55E–12	6.48E–08	0.00%
^{211}Po	3.35E–12	1.8	6.04E–12	2.02E–09	^{211}Po	3.35E–12	2.88E–06	9.66E–18	4.82E–14	2.02E–09	0.00%
^{212}Po	4.60E–08	1.8	8.29E–08	2.77E–05	^{212}Po	4.60E–08	2.88E–06	1.33E–13	6.61E–10	2.77E–05	0.00%
^{213}Po	5.49E–12	1.8	9.88E–12	3.30E–09	^{213}Po	5.49E–12	2.88E–06	1.58E–17	7.89E–14	3.30E–09	0.00%
^{214}Po	2.87E–10	1.8	5.17E–10	1.73E–07	^{214}Po	2.87E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%
^{215}Po	1.20E–09	1.8	2.16E–09	7.21E–07	^{215}Po	1.20E–09	2.88E–06	3.45E–15	1.72E–11	7.21E–07	0.00%
^{216}Po	7.18E–08	1.8	1.29E–07	4.32E–05	^{216}Po	7.18E–08	2.88E–06	2.07E–13	1.03E–09	4.32E–05	0.00%
^{218}Po	2.88E–10	1.8	5.18E–10	1.73E–07	^{218}Po	2.88E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%
^{144}Pr	2.33E–10	1.8	4.19E–10	1.40E–07	^{144}Pr	2.33E–10	2.88E–06	6.70E–16	3.34E–12	1.40E–07	0.00%
^{144m}Pr	2.79E–12	1.8	5.03E–12	1.68E–09	^{144m}Pr	2.79E–12	2.88E–06	8.04E–18	4.01E–14	1.68E–09	0.00%
^{236}Pu	6.52E–09	1.8	1.17E–08	3.93E–06	^{236}Pu	6.52E–09	2.88E–06	1.88E–14	9.37E–11	3.93E–06	0.00%
^{238}Pu	[d]		9.23E–03	3.09E+00	^{238}Pu	[e]		5.85E–08	2.92E–04	3.09E+00	0.48%
^{239}Pu	[d]		2.75E–03	9.20E–01	^{239}Pu	[e]		9.36E–09	4.67E–05	9.20E–01	0.14%
^{240}Pu	6.06E–04	1.8	1.09E–03	3.65E–01	^{240}Pu	6.06E–04	2.88E–06	1.75E–09	8.71E–06	3.65E–01	0.06%
^{241}Pu	8.75E–03	1.8	1.58E–02	5.27E+00	^{241}Pu	[e]		3.75E–08	1.87E–04	5.27E+00	0.82%
^{242}Pu	4.43E–07	1.8	7.98E–07	2.67E–04	^{242}Pu	4.43E–07	2.88E–06	1.28E–12	6.37E–09	2.67E–04	0.00%
^{243}Pu	1.22E–16	1.8	2.20E–16	7.34E–14	^{243}Pu	1.22E–16	2.88E–06	3.51E–22	1.75E–18	7.34E–14	0.00%
^{244}Pu	2.94E–14	1.8	5.29E–14	1.77E–11	^{244}Pu	2.94E–14	2.88E–06	8.47E–20	4.22E–16	1.77E–11	0.00%
^{223}Ra	1.20E–09	1.8	2.16E–09	7.21E–07	^{223}Ra	1.20E–09	2.88E–06	3.45E–15	1.72E–11	7.21E–07	0.00%
^{224}Ra	7.18E–08	1.8	1.29E–07	4.32E–05	^{224}Ra	7.18E–08	2.88E–06	2.07E–13	1.03E–09	4.32E–05	0.00%
^{225}Ra	5.61E–12	1.8	1.01E–11	3.37E–09	^{225}Ra	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{226}Ra	2.88E–10	1.8	5.18E–10	1.73E–07	^{226}Ra	2.88E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%

Table A-11. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{228}Ra	1.73E–14	1.8	3.11E–14	1.04E–11	^{228}Ra	1.73E–14	2.88E–06	4.97E–20	2.48E–16	1.04E–11	0.00%
^{87}Rb	7.15E–10	1.8	1.29E–09	4.30E–07	^{87}Rb	7.15E–10	2.88E–06	2.06E–15	1.03E–11	4.30E–07	0.00%
^{102}Rh	2.46E–09	1.8	4.44E–09	1.48E–06	^{102}Rh	2.46E–09	2.88E–06	7.10E–15	3.54E–11	1.48E–06	0.00%
^{106}Rh	7.69E–09	1.8	1.38E–08	4.63E–06	^{106}Rh	7.69E–09	2.88E–06	2.21E–14	1.10E–10	4.63E–06	0.00%
^{219}Rn	1.20E–09	1.8	2.16E–09	7.21E–07	^{219}Rn	1.20E–09	2.88E–06	3.45E–15	1.72E–11	7.21E–07	0.00%
^{220}Rn	7.18E–08	1.8	1.29E–07	4.32E–05	^{220}Rn	7.18E–08	2.88E–06	2.07E–13	1.03E–09	4.32E–05	0.00%
^{222}Rn	2.88E–10	1.8	5.18E–10	1.73E–07	^{222}Rn	2.88E–10	2.88E–06	8.28E–16	4.13E–12	1.73E–07	0.00%
^{106}Ru	7.69E–09	1.8	1.38E–08	4.63E–06	^{106}Ru	7.69E–09	2.88E–06	2.21E–14	1.10E–10	4.63E–06	0.00%
^{125}Sb	[d]		5.55E–04	1.86E–01	^{125}Sb	[e]		1.95E–09	9.73E–06	1.86E–01	0.03%
^{126}Sb	1.41E–06	1.8	2.55E–06	8.51E–04	^{126}Sb	1.41E–06	2.88E–06	4.07E–12	2.03E–08	8.51E–04	0.00%
^{126m}Sb	1.01E–05	1.8	1.82E–05	6.08E–03	^{126m}Sb	1.01E–05	2.88E–06	2.91E–11	1.45E–07	6.08E–03	0.00%
^{79}Se	1.07E–05	1.8	1.93E–05	6.46E–03	^{79}Se	1.07E–05	2.88E–06	3.09E–11	1.54E–07	6.46E–03	0.00%
^{146}Sm	4.65E–12	1.8	8.36E–12	2.80E–09	^{146}Sm	4.65E–12	2.88E–06	1.34E–17	6.67E–14	2.80E–09	0.00%
^{147}Sm	1.81E–10	1.8	3.26E–10	1.09E–07	^{147}Sm	1.81E–10	2.88E–06	5.22E–16	2.60E–12	1.09E–07	0.00%
^{148}Sm	9.30E–16	1.8	1.67E–15	5.60E–13	^{148}Sm	9.30E–16	2.88E–06	2.68E–21	1.34E–17	5.60E–13	0.00%
^{149}Sm	8.26E–17	1.8	1.49E–16	4.97E–14	^{149}Sm	8.26E–17	2.88E–06	2.38E–22	1.19E–18	4.97E–14	0.00%
^{151}Sm	7.71E–03	1.8	1.39E–02	4.64E+00	^{151}Sm	7.71E–03	2.88E–06	2.22E–08	1.11E–04	4.64E+00	0.72%
^{119m}Sn	9.55E–15	1.8	1.72E–14	5.75E–12	^{119m}Sn	9.55E–15	2.88E–06	2.75E–20	1.37E–16	5.75E–12	0.00%
^{121m}Sn	5.74E–05	1.8	1.03E–04	3.45E–02	^{121m}Sn	5.74E–05	2.88E–06	1.65E–10	8.24E–07	3.45E–02	0.01%
^{126}Sn	1.01E–05	1.8	1.82E–05	6.08E–03	^{126}Sn	1.01E–05	2.88E–06	2.91E–11	1.45E–07	6.08E–03	0.00%
^{90}Sr	[d]		1.87E–02	6.25E+00	^{90}Sr	[e]		1.83E–06	9.13E–03	6.26E+00	0.97%
^{98}Tc	6.34E–11	1.8	1.14E–10	3.82E–08	^{98}Tc	6.34E–11	2.88E–06	1.83E–16	9.12E–13	3.82E–08	0.00%
^{99}Tc	[d]		6.17E–04	2.06E–01	^{99}Tc	[e]		1.35E–09	6.74E–06	2.06E–01	0.03%
^{123}Te	5.12E–17	1.8	9.22E–17	3.08E–14	^{123}Te	5.12E–17	2.88E–06	1.48E–22	7.36E–19	3.08E–14	0.00%
^{125m}Te	1.43E–05		1.36E–04	4.55E–02	^{125m}Te	1.43E–05	2.88E–06	4.13E–11	2.06E–07	4.55E–02	0.01%

Table A-11. (continued).

Solids					Liquids					Total (Solids+Liquids)	
Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor[c]	(Ci/kg)	Total Solids Activity (Ci)	Nuclide	ORIGEN2[a,b]	^{137}Cs Ratio Factor	Ci/L	Total Liquids Activity (Ci)	Total Tank Activity (Ci)	Percent Total Activity
^{227}Th	1.18E–09	1.8	2.13E–09	7.11E–07	^{227}Th	1.18E–09	2.88E–06	3.40E–15	1.70E–11	7.11E–07	0.00%
^{228}Th	7.16E–08	1.8	1.29E–07	4.31E–05	^{228}Th	7.16E–08	2.88E–06	2.06E–13	1.03E–09	4.31E–05	0.00%
^{229}Th	5.61E–12	1.8	1.01E–11	3.37E–09	^{229}Th	5.61E–12	2.88E–06	1.61E–17	8.06E–14	3.37E–09	0.00%
^{230}Th	2.35E–08	1.8	4.24E–08	1.42E–05	^{230}Th	2.35E–08	2.88E–06	6.78E–14	3.38E–10	1.42E–05	0.00%
^{231}Th	7.93E–07	1.8	1.43E–06	4.77E–04	^{231}Th	7.93E–07	2.88E–06	2.28E–12	1.14E–08	4.77E–04	0.00%
^{232}Th	1.85E–14	1.8	3.33E–14	1.11E–11	^{232}Th	1.85E–14	2.88E–06	5.33E–20	2.66E–16	1.11E–11	0.00%
^{234}Th	8.17E–07	1.8	1.47E–06	4.92E–04	^{234}Th	8.17E–07	2.88E–06	2.35E–12	1.17E–08	4.92E–04	0.00%
^{207}Tl	1.19E–09	1.8	2.15E–09	7.19E–07	^{207}Tl	1.19E–09	2.88E–06	3.44E–15	1.72E–11	7.19E–07	0.00%
^{208}Tl	2.58E–08	1.8	4.65E–08	1.55E–05	^{208}Tl	2.58E–08	2.88E–06	7.43E–14	3.71E–10	1.55E–05	0.00%
^{209}Tl	1.21E–13	1.8	2.18E–13	7.29E–11	^{209}Tl	1.21E–13	2.88E–06	3.49E–19	1.74E–15	7.29E–11	0.00%
^{171}Tm	4.80E–16	1.8	8.63E–16	2.89E–13	^{171}Tm	4.80E–16	2.88E–06	1.38E–21	6.89E–18	2.89E–13	0.00%
^{232}U	6.91E–08	1.8	1.24E–07	4.16E–05	^{232}U	6.91E–08	2.88E–06	1.99E–13	9.93E–10	4.16E–05	0.00%
^{233}U	1.77E–09	1.8	3.19E–09	1.07E–06	^{233}U	1.77E–09	2.88E–06	5.11E–15	2.55E–11	1.07E–06	0.00%
^{234}U	[d]		2.98E–06	9.96E–04	^{234}U	3.37E–05	2.88E–06	9.71E–11	4.84E–07	9.97E–04	0.00%
^{235}U	7.93E–07	1.8	1.43E–06	4.77E–04	^{235}U	7.93E–07	2.88E–06	2.28E–12	1.14E–08	4.77E–04	0.00%
^{236}U	1.85E–06	1.8	3.33E–06	1.11E–03	^{236}U	1.85E–06	2.88E–06	5.33E–12	2.66E–08	1.11E–03	0.00%
^{237}U	2.15E–07	1.8	3.86E–07	1.29E–04	^{237}U	2.15E–07	2.88E–06	6.18E–13	3.08E–09	1.29E–04	0.00%
^{238}U	8.17E–07	1.8	1.47E–06	4.92E–04	^{238}U	8.17E–07	2.88E–06	2.35E–12	1.17E–08	4.92E–04	0.00%
^{240}U	2.94E–14	1.8	5.28E–14	1.77E–11	^{240}U	2.94E–14	2.88E–06	8.46E–20	4.22E–16	1.77E–11	0.00%
^{90}Y	[d]		1.87E–02	6.25E+00	^{90}Y	8.88E–01	2.88E–06	2.56E–06	1.28E–02	6.27E+00	0.97%
^{93}Zr	1.06E–04	1.8	1.90E–04	6.36E–02	^{93}Zr	1.06E–04	2.88E–06	3.04E–10	1.52E–06	6.36E–02	0.01%
			Total	645.8				Total	0.05	645.8	

a. Source: Wenzel 2005 (reports nuclide to ^{137}Cs ratios based on ORIGEN2 modeling and nuclide to ^{137}Cs ratios calculated from past analytical data).
b. From decay of parent sample data to 2012.
c. Average of ^{137}Cs data collected from the 1999 sampling of Tank WM-188.
d. Measured solid sample value from WM-183.
e. Measured liquid sample from Tank WM-186.

A-2. COMPARISION OF THE WM-182 POST-DECONTAMINATION RADIONUCLIDE INVENTORY TO THE PA INVENTORY

Results of post-decontamination sampling and analysis were used to determine the concentrations of the radioactive and hazardous constituents remaining in each tank. Analysis results were used to confirm that the radionuclide concentrations met the closure requirements, and that they were bounded by the concentrations assumed in the conservative inventory in the 2003 PA. Table A-12 presents a comparison between the estimated inventory at closure for Tank WM-182 and the conservative PA inventory for a single tank. The PA inventory was decayed to 2016 and the inventory at closure was decayed to 2012; because of this, it may be difficult to directly compare results for short-lived radionuclides. However, the total Ci remaining in the tank at closure are an order of magnitude lower than the amount estimated by the conservative PA inventory.

Table A-12. Comparison of post-decontamination estimated inventory for WM-182 and the performance assessment inventory.

	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
Nuclide	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{225}Ac	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{227}Ac	5.17E–06	8.58E–07	6.03E–06	0.00%	2.66E–06	1.33E–09	2.66E–06	0.00%	0.44
^{228}Ac	6.19E–11	1.03E–11	7.22E–11	0.00%	3.85E–11	1.92E–14	3.85E–11	0.00%	0.53
^{108}Ag	3.79E–08	6.29E–09	4.42E–08	0.00%	6.35E–10	3.17E–13	6.35E–10	0.00%	0.01
^{108m}Ag	0.00E+00	0.00E+00	0.00E+00	0.00%	7.13E–09	3.56E–12	7.13E–09	0.00%	NA
^{109m}Ag	0.00E+00	0.00E+00	0.00E+00	0.00%	5.30E–14	2.65E–17	5.30E–14	0.00%	NA
^{110}Ag	0.00E+00	0.00E+00	0.00E+00	0.00%	1.99E–15	9.93E–19	1.99E–15	0.00%	NA
^{110m}Ag	0.00E+00	0.00E+00	0.00E+00	0.00%	1.50E–13	7.47E–17	1.50E–13	0.00%	NA
^{241}Am	4.38E–01	1.79E–01	6.17E–01	0.00%	4.21E–01	5.30E–04	4.22E–01	0.02%	0.68
^{242}Am	1.34E–03	2.23E–04	1.56E–03	0.00%	1.59E–03	7.94E–07	1.59E–03	0.00%	1.02
^{242m}Am	1.34E–03	2.23E–04	1.56E–03	0.00%	1.60E–03	7.98E–07	1.60E–03	0.00%	1.02
^{243}Am	1.93E–03	3.19E–04	2.25E–03	0.00%	2.19E–03	1.09E–06	2.19E–03	0.00%	0.97
^{217}At	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{137m}Ba	3.44E+03	5.71E+02	4.01E+03	16.64%	1.14E+03	1.11E+00	1.14E+03	47.63%	0.28
^{10}Be	2.75E–07	4.56E–08	3.21E–07	0.00%	1.69E–07	8.41E–11	1.69E–07	0.00%	0.53
^{210}Bi	5.17E–07	8.58E–08	6.03E–07	0.00%	2.48E–07	1.24E–10	2.48E–07	0.00%	0.41
^{210m}Bi	2.00E–20	3.32E–21	2.33E–20	0.00%	1.21E–20	6.02E–24	1.21E–20	0.00%	0.52
^{211}Bi	5.17E–06	8.58E–07	6.03E–06	0.00%	2.67E–06	1.33E–09	2.67E–06	0.00%	0.44
^{212}Bi	1.72E–04	2.87E–05	2.01E–04	0.00%	1.60E–04	7.99E–08	1.60E–04	0.00%	0.80
^{213}Bi	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{214}Bi	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{14}C	1.10E–05	4.94E–01	4.94E–01	0.00%	4.90E–06	5.39E–08	4.95E–06	0.00%	0.00
^{109}Cd	0.00E+00	0.00E+00	0.00E+00	0.00%	5.30E–14	2.65E–17	5.30E–14	0.00%	NA
^{113m}Cd	1.65E–01	2.74E–02	1.92E–01	0.00%	1.29E–01	6.43E–05	1.29E–01	0.01%	0.67
^{142}Ce	2.75E–06	4.56E–07	3.21E–06	0.00%	1.63E–06	8.14E–10	1.63E–06	0.00%	0.51
^{144}Ce	5.50E–07	9.13E–08	6.41E–07	0.00%	5.18E–07	2.59E–10	5.18E–07	0.00%	0.81
^{249}Cf	1.55E–12	2.57E–13	1.81E–12	0.00%	1.61E–12	8.02E–16	1.61E–12	0.00%	0.89

Table A-12. (continued).

Nuclide	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{250}Cf	6.55E–13	1.09E–13	7.64E–13	0.00%	8.31E–13	4.15E–16	8.31E–13	0.00%	1.09
^{251}Cf	2.44E–14	4.07E–15	2.85E–14	0.00%	2.55E–14	1.27E–17	2.55E–14	0.00%	0.90
^{252}Cf	0.00E+00	0.00E+00	0.00E+00	0.00%	1.08E–15	5.39E–19	1.08E–15	0.00%	NA
^{242}Cm	1.10E–03	1.83E–04	1.28E–03	0.00%	1.32E–03	6.58E–07	1.32E–03	0.00%	1.03
^{243}Cm	1.93E–03	3.19E–04	2.25E–03	0.00%	2.86E–04	1.43E–07	2.86E–04	0.00%	0.13
^{244}Cm	9.63E–02	1.60E–02	1.12E–01	0.00%	1.57E–02	7.82E–06	1.57E–02	0.00%	0.14
^{245}Cm	2.75E–05	4.56E–06	3.21E–05	0.00%	3.74E–06	1.87E–09	3.74E–06	0.00%	0.12
^{246}Cm	1.79E–06	2.97E–07	2.09E–06	0.00%	2.45E–07	1.23E–10	2.45E–07	0.00%	0.12
^{247}Cm	2.00E–12	3.32E–13	2.33E–12	0.00%	2.72E–13	1.36E–16	2.72E–13	0.00%	0.12
^{248}Cm	2.17E–12	3.59E–13	2.53E–12	0.00%	2.87E–13	1.43E–16	2.87E–13	0.00%	0.11
^{60}Co	1.41E–01	6.96E–02	2.11E–01	0.00%	6.21E–02	4.12E–04	6.25E–02	0.00%	0.30
^{134}Cs	4.71E–02	6.04E–03	5.31E–02	0.00%	6.76E–02	3.37E–05	6.76E–02	0.00%	1.27
^{135}Cs	8.26E–02	1.37E–02	9.63E–02	0.00%	3.20E–02	1.60E–05	3.20E–02	0.00%	0.33
^{137}Cs	3.44E+03	5.71E+02	4.01E+03	16.64%	1.14E+03	1.11E+00	1.14E+03	47.63%	0.28
^{150}Eu	1.03E–06	1.71E–07	1.20E–06	0.00%	6.61E–07	3.30E–10	6.61E–07	0.00%	0.55
^{152}Eu	1.21E–01	2.00E–02	1.41E–01	0.00%	8.73E–02	4.36E–05	8.73E–02	0.00%	0.62
^{154}Eu	3.00E–01	9.10E–01	1.21E+00	0.01%	3.89E+00	3.19E–04	3.89E+00	0.16%	3.22
^{155}Eu	2.44E+00	1.12E–01	2.55E+00	0.01%	1.06E+00	5.27E–04	1.06E+00	0.04%	0.42
^{55}Fe	0.00E+00	0.00E+00	0.00E+00	0.00%	1.09E+00	5.43E–04	1.09E+00	0.05%	NA
^{221}Fr	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{223}Fr	7.23E–08	1.20E–08	8.43E–08	0.00%	3.68E–08	1.84E–11	3.68E–08	0.00%	0.44
^{152}Gd	1.34E–13	2.23E–14	1.56E–13	0.00%	7.97E–14	3.98E–17	7.97E–14	0.00%	0.51
^{153}Gd	0.00E+00	0.00E+00	0.00E+00	0.00%	9.26E–16	4.62E–19	9.26E–16	0.00%	NA
^{3}H	4.82E–01	8.01E–02	5.62E–01	0.00%	7.17E–01	1.66E–05	7.17E–01	0.03%	1.28
^{166m}Ho	4.13E–06	6.86E–07	4.82E–06	0.00%	2.51E–06	1.25E–09	2.51E–06	0.00%	0.52
^{129}I	2.24E–03[a]	3.71E–04[b]	2.61E–03	0.00%	7.73E–04	1.12E–06	7.74E–04	0.00%	0.30
^{115}In	8.94E–12	1.48E–12	1.04E–11	0.00%	6.13E–13	3.06E–16	6.13E–13	0.00%	0.06

Table A-12. (continued).

Nuclide	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{138}La	1.79E–11	2.97E–12	2.09E–11	0.00%	1.04E–11	5.21E–15	1.04E–11	0.00%	0.50
^{93m}Nb	1.72E–01	2.87E–02	2.01E–01	0.00%	1.95E–01	9.72E–05	1.95E–01	0.01%	0.97
^{94}Nb	7.67E+00	1.71E–02	7.69E+00	0.03%	2.06E–01	4.03E–05	2.06E–01	0.01%	0.03
^{144}Nd	1.48E–10	2.46E–11	1.73E–10	0.00%	8.83E–11	4.41E–14	8.83E–11	0.00%	0.51
^{59}Ni	3.68E–02[a]	6.11E–03[b]	4.29E–02	0.00%	2.51E–02	1.25E–05	2.51E–02	0.00%	0.59
^{63}Ni	2.61E+00	4.34E–01	3.04E+00	0.01%	2.86E+00	1.43E–03	2.86E+00	0.12%	0.94
^{236}Np	0.00E+00	0.00E+00	0.00E+00	0.00%	4.91E–07	2.45E–10	4.91E–07	0.00%	NA
^{237}Np	5.92E–03	1.71E–03	7.63E–03	0.00%	4.70E–02	2.71E–07	4.70E–02	0.00%	6.16
^{238}Np	6.55E–06	1.09E–06	7.64E–06	0.00%	7.99E–06	3.99E–09	7.99E–06	0.00%	1.05
^{239}Np	1.93E–03	3.19E–04	2.25E–03	0.00%	2.19E–03	1.09E–06	2.19E–03	0.00%	0.97
^{240m}Np	6.19E–11	1.03E–11	7.22E–11	0.00%	6.54E–11	3.27E–14	6.54E–11	0.00%	0.91
^{231}Pa	8.94E–06	1.48E–06	1.04E–05	0.00%	4.73E–06	2.36E–09	4.73E–06	0.00%	0.45
^{233}Pa	2.69E–01	4.47E–02	3.14E–01	0.00%	4.70E–02	2.35E–05	4.70E–02	0.00%	0.15
^{234}Pa	2.44E–06	4.07E–07	2.85E–06	0.00%	2.37E–06	1.18E–09	2.37E–06	0.00%	0.83
^{234m}Pa	1.93E–03	3.19E–04	2.25E–03	0.00%	1.82E–03	9.09E–07	1.82E–03	0.00%	0.81
^{209}Pb	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{210}Pb	5.17E–07	8.58E–08	6.03E–07	0.00%	2.48E–07	1.24E–10	2.48E–07	0.00%	0.41
^{211}Pb	5.17E–06	8.58E–07	6.03E–06	0.00%	2.67E–06	1.33E–09	2.67E–06	0.00%	0.44
^{212}Pb	1.79E–04	2.97E–05	2.09E–04	0.00%	1.60E–04	7.99E–08	1.60E–04	0.00%	0.77
^{214}Pb	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{107}Pd	1.48E–03	2.46E–04	1.73E–03	0.00%	9.05E–04	4.52E–07	9.05E–04	0.00%	0.52
^{146}Pm	8.94E–04	1.48E–04	1.04E–03	0.00%	8.91E–04	4.45E–07	8.91E–04	0.00%	0.86
^{147}Pm	5.17E–01	8.58E–02	6.03E–01	0.00%	8.65E–01	4.32E–04	8.65E–01	0.04%	1.44
^{210}Po	5.17E–07	8.58E–08	6.03E–07	0.00%	2.40E–07	1.20E–10	2.40E–07	0.00%	0.40
^{211}Po	0.00E+00	0.00E+00	0.00E+00	0.00%	7.47E–09	3.73E–12	7.47E–09	0.00%	NA
^{212}Po	1.10E–04	1.83E–05	1.28E–04	0.00%	1.03E–04	5.12E–08	1.03E–04	0.00%	0.80
^{213}Po	3.44E–08	5.71E–09	4.01E–08	0.00%	1.22E–08	6.11E–12	1.22E–08	0.00%	0.30

Table A-12. (continued).

Nuclide	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{214}Po	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{215}Po	5.17E–06	8.58E–07	6.03E–06	0.00%	2.67E–06	1.33E–09	2.67E–06	0.00%	0.44
^{216}Po	1.79E–04	2.97E–05	2.09E–04	0.00%	1.60E–04	7.99E–08	1.60E–04	0.00%	0.77
^{218}Po	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{144}Pr	5.50E–07	9.13E–08	6.41E–07	0.00%	5.18E–07	2.59E–10	5.18E–07	0.00%	0.81
^{144m}Pr	6.55E–09	1.09E–09	7.64E–09	0.00%	6.22E–09	3.11E–12	6.22E–09	0.00%	0.81
^{236}Pu	1.03E–05	1.71E–06	1.20E–05	0.00%	1.45E–05	7.26E–09	1.45E–05	0.00%	1.21
^{238}Pu	1.41E+01	2.84E+00	1.69E+01	0.07%	1.14E+01	2.47E–03	1.14E+01	0.48%	0.67
^{239}Pu	8.97E–01	3.52E–01	1.25E+00	0.01%	3.40E+00	2.44E–04	3.40E+00	0.14%	2.72
^{240}Pu	9.63E–01	1.60E–01	1.12E+00	0.00%	1.35E+00	6.74E–04	1.35E+00	0.06%	1.20
^{241}Pu	1.27E+01	2.11E+00	1.48E+01	0.06%	1.95E+01	5.89E–04	1.95E+01	0.81%	1.32
^{242}Pu	7.23E–04	1.20E–04	8.43E–04	0.00%	9.87E–04	4.93E–07	9.87E–04	0.00%	1.17
^{243}Pu	0.00E+00	0.00E+00	0.00E+00	0.00%	2.72E–13	1.36E–16	2.72E–13	0.00%	NA
^{244}Pu	6.19E–11	1.03E–11	7.22E–11	0.00%	6.55E–11	3.27E–14	6.55E–11	0.00%	0.91
^{223}Ra	5.17E–06	8.58E–07	6.03E–06	0.00%	2.67E–06	1.33E–09	2.67E–06	0.00%	0.44
^{224}Ra	1.79E–04	2.97E–05	2.09E–04	0.00%	1.60E–04	7.99E–08	1.60E–04	0.00%	0.77
^{225}Ra	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{226}Ra	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{228}Ra	6.19E–11	1.03E–11	7.22E–11	0.00%	3.85E–11	1.92E–14	3.85E–11	0.00%	0.53
^{87}Rb	2.69E–06	4.47E–07	3.14E–06	0.00%	1.59E–06	7.95E–10	1.59E–06	0.00%	0.51
^{102}Rh	3.44E–06	5.71E–07	4.01E–06	0.00%	5.49E–06	2.74E–09	5.49E–06	0.00%	1.37
^{106}Rh	1.10E–05	1.83E–06	1.28E–05	0.00%	1.71E–05	8.55E–09	1.71E–05	0.00%	1.33
^{219}Rn	5.17E–06	8.58E–07	6.03E–06	0.00%	2.67E–06	1.33E–09	2.67E–06	0.00%	0.44
^{220}Rn	1.79E–04	2.97E–05	2.09E–04	0.00%	1.60E–04	7.99E–08	1.60E–04	0.00%	0.77
^{222}Rn	1.21E–06	2.00E–07	1.41E–06	0.00%	6.41E–07	3.20E–10	6.41E–07	0.00%	0.45
^{106}Ru	1.10E–05	1.83E–06	1.28E–05	0.00%	1.71E–05	8.55E–09	1.71E–05	0.00%	1.33
^{125}Sb	4.48E–02	7.43E–03	5.22E–02	0.00%	6.87E–01	1.76E–03	6.89E–01	0.03%	13.19

Table A-12. (continued).

Nuclide	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{126}Sb	5.17E–03	8.58E–04	6.03E–03	0.00%	3.15E–03	1.57E–06	3.15E–03	0.00%	0.52
^{126m}Sb	3.79E–02	6.29E–03	4.42E–02	0.00%	2.25E–02	1.12E–05	2.25E–02	0.00%	0.51
^{79}Se	4.13E–02	6.86E–03	4.82E–02	0.00%	2.39E–02	1.19E–05	2.39E–02	0.00%	0.50
^{146}Sm	2.54E–08	4.22E–09	2.96E–08	0.00%	1.04E–08	5.17E–12	1.04E–08	0.00%	0.35
^{147}Sm	6.88E–07	1.14E–07	8.02E–07	0.00%	4.04E–07	2.02E–10	4.04E–07	0.00%	0.50
^{148}Sm	3.44E–12	5.71E–13	4.01E–12	0.00%	2.07E–12	1.03E–15	2.07E–12	0.00%	0.52
^{149}Sm	3.13E–13	5.19E–14	3.65E–13	0.00%	1.84E–13	9.19E–17	1.84E–13	0.00%	0.50
^{151}Sm	2.81E+01	4.69E+00	3.28E+01	0.14%	1.72E+01	8.57E–03	1.72E+01	0.72%	0.52
^{119m}Sn	0.00E+00	0.00E+00	0.00E+00	0.00%	2.13E–11	1.06E–14	2.13E–11	0.00%	NA
^{121m}Sn	4.47E–03	7.43E–04	5.21E–03	0.00%	1.28E–01	6.38E–05	1.28E–01	0.01%	24.57
^{126}Sn	3.79E–02	6.29E–03	4.42E–02	0.00%	2.25E–02	1.12E–05	2.25E–02	0.00%	0.51
^{90}Sr	7.59E+03	4.07E+02	8.00E+03	33.18%	2.32E+01	2.41E–01	2.34E+01	0.98%	0.00
^{98}Tc	2.38E–07	3.94E–08	2.77E–07	0.00%	1.41E–07	7.06E–11	1.41E–07	0.00%	0.51
^{99}Tc	8.97E–01[a]	1.49E–01[b]	1.05E+00	0.00%	7.64E–01	4.54E–05	7.64E–01	0.03%	0.73
^{123}Te	3.44E–14	5.71E–15	4.01E–14	0.00%	1.14E–13	5.70E–17	1.14E–13	0.00%	2.84
^{125m}Te	1.14E–02	1.89E–03	1.33E–02	0.00%	1.68E–01	1.59E–05	1.68E–01	0.01%	12.64
^{227}Th	5.17E–06	8.58E–07	6.03E–06	0.00%	2.63E–06	1.31E–09	2.63E–06	0.00%	0.44
^{228}Th	1.79E–04	2.97E–05	2.09E–04	0.00%	1.60E–04	7.97E–08	1.60E–04	0.00%	0.77
^{229}Th	3.44E–08	5.71E–09	4.01E–08	0.00%	1.25E–08	6.24E–12	1.25E–08	0.00%	0.31
^{230}Th	8.26E–05	1.37E–05	9.63E–05	0.00%	5.24E–05	2.62E–08	5.24E–05	0.00%	0.54
^{231}Th	1.93E–03	3.19E–04	2.25E–03	0.00%	1.77E–03	8.82E–07	1.77E–03	0.00%	0.79
^{232}Th	6.55E–11	1.09E–11	7.64E–11	0.00%	4.12E–11	2.06E–14	4.12E–11	0.00%	0.54
^{234}Th	1.93E–03	3.19E–04	2.25E–03	0.00%	1.82E–03	9.09E–07	1.82E–03	0.00%	0.81
^{207}Tl	5.17E–06	8.58E–07	6.03E–06	0.00%	2.66E–06	1.33E–09	2.66E–06	0.00%	0.44
^{208}Tl	6.19E–05	1.03E–05	7.22E–05	0.00%	5.75E–05	2.87E–08	5.75E–05	0.00%	0.80
^{209}Tl	7.57E–10	1.26E–10	8.83E–10	0.00%	2.70E–10	1.35E–13	2.70E–10	0.00%	0.31
^{171}Tm	4.13E–13	6.86E–14	4.82E–13	0.00%	1.07E–12	5.34E–16	1.07E–12	0.00%	2.22

Table A-12. (continued).

Nuclide	Performance Assessment Conservative Inventory decayed to 2016 (Ci)				WM-182 Conservative Inventory decayed to 2012 (Ci)				Ratio WM-182 to PA
	Solids	Liquids	Total	Percent of Total	Solids	Liquids	Total	Percent of Total	
^{232}U	1.72E–04	2.87E–05	2.01E–04	0.00%	1.54E–04	7.69E–08	1.54E–04	0.00%	0.77
^{233}U	2.23E–05	3.72E–06	2.60E–05	0.00%	3.95E–06	1.97E–09	3.95E–06	0.00%	0.15
^{234}U	7.57E–02	1.26E–02	8.83E–02	0.00%	3.69E–03	3.08E–06	3.69E–03	0.00%	0.04
^{235}U	4.11E–04	5.99E–05	4.71E–04	0.00%	1.77E–03	8.82E–07	1.77E–03	0.00%	3.76
^{236}U	3.06E–03	3.24E–05	3.09E–03	0.00%	4.12E–03	2.06E–06	4.12E–03	0.00%	1.33
^{237}U	3.19E–04	5.31E–05	3.72E–04	0.00%	4.78E–04	2.39E–07	4.78E–04	0.00%	1.29
^{238}U	2.44E–04	8.18E–05	3.26E–04	0.00%	1.82E–03	9.09E–07	1.82E–03	0.00%	5.59
^{240}U	6.19E–11	1.03E–11	7.22E–11	0.00%	6.54E–11	3.27E–14	6.54E–11	0.00%	0.91
^{90}Y	7.59E+03	4.07E+02	8.00E+03	33.18%	2.32E+01	2.41E–01	2.34E+01	0.98%	0.00
^{93}Zr	2.00E–01	3.32E–02	2.33E–01	0.00%	2.36E–01	1.18E–04	2.36E–01	0.01%	1.01
Total	22,133	1,969	24,103		2,393	3	2,396		0.10

NA=Not Applicable. Data for this radionuclide were not available in the PA inventory; therefore, a ratio cannot be calculated.
a. Measured liquid sample from Tank WM-186.
b. Measured solid sample value from WM-183.

A-3. POST-DECONTAMINATION INTERIOR TANK PHOTOGRAPHS FOR 300,000-GAL TANKS

Photographs of the 300,000-gal tanks were taken before and after decontamination to provide additional information on the extent of the decontamination process. Post-decontamination photos show that only isolated areas of solid residuals remain on the floors of the tanks (see Figures A-1 through A-7). Photographs of the 30,000-gal tanks are not included since photograph-quality cameras and lights could not be placed in the tanks as part of the video inspection.

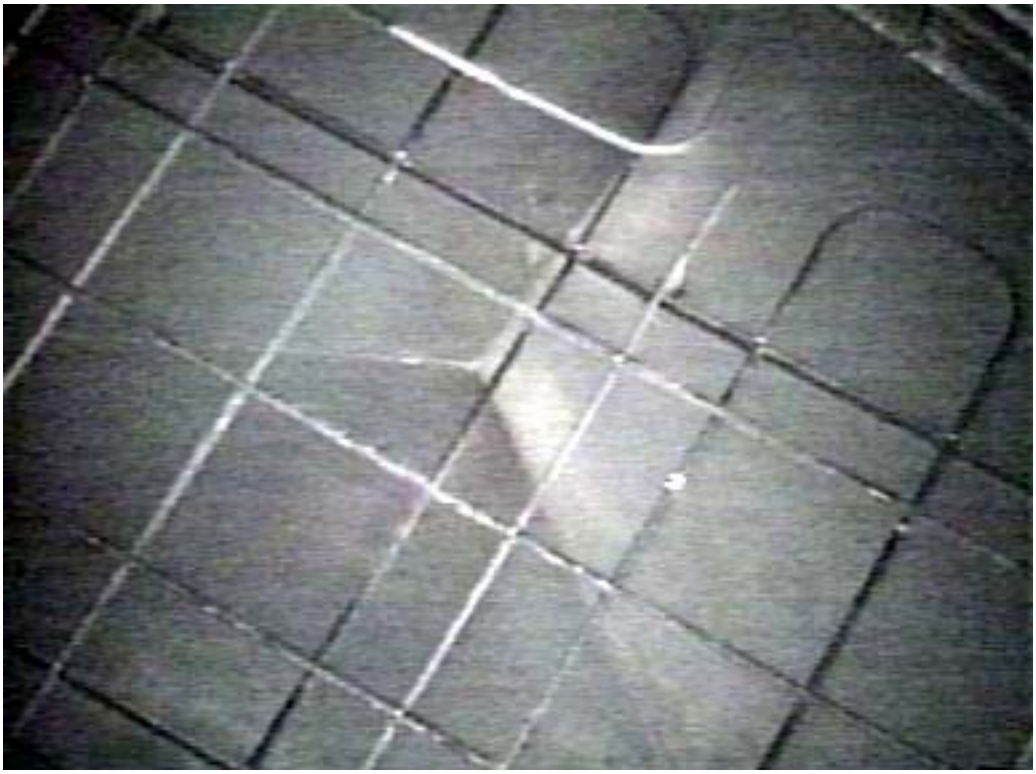

Figure A-1. TFF Tank WM-180 post-decontamination interior.

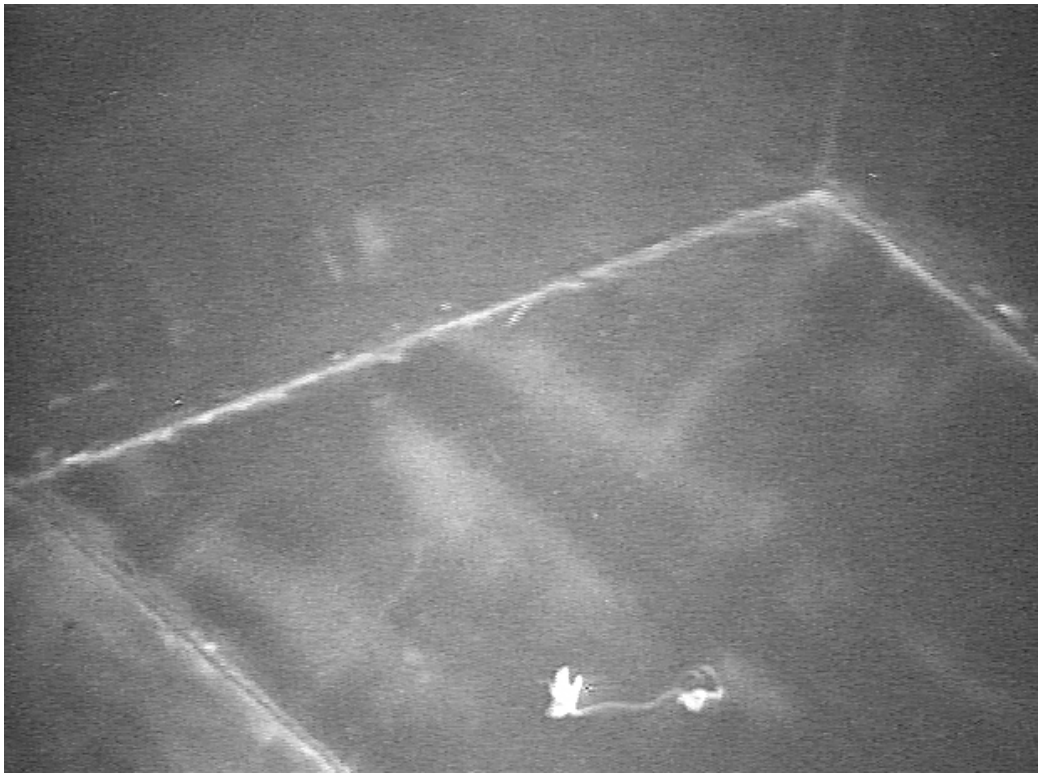

Figure A-2. TFF Tank WM-181 post-decontamination interior.

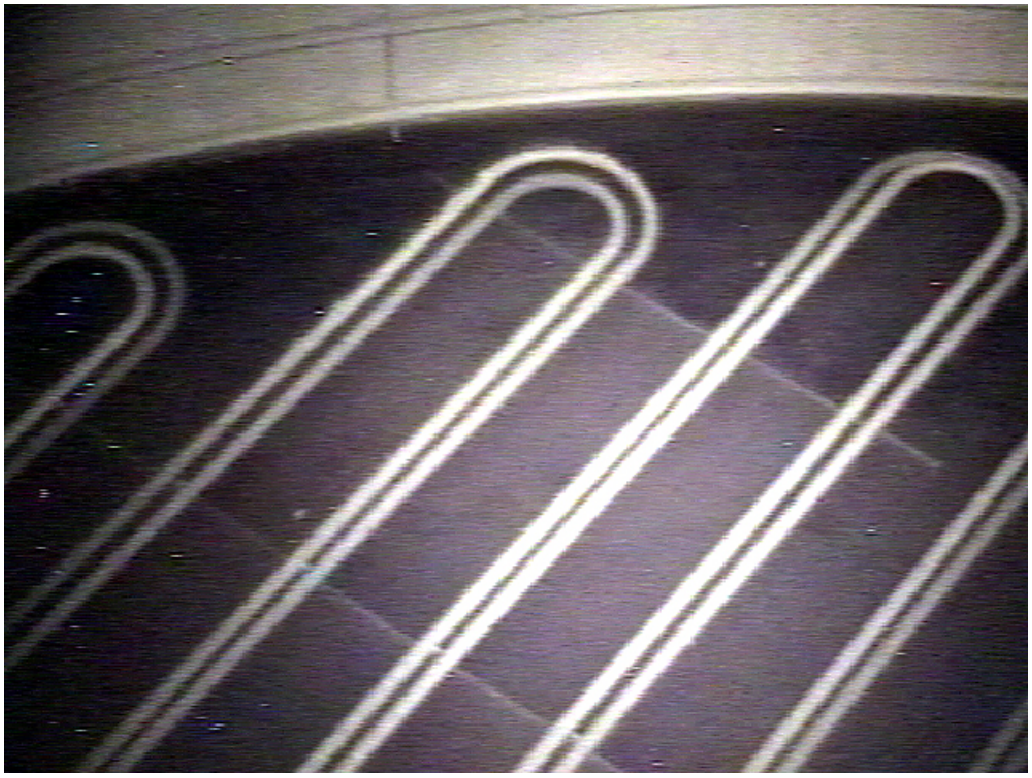

Figure A-3. TFF Tank WM-182 post-decontamination interior.

Figure A-4. TFF Tank WM-183 post-decontamination interior.

Figure A-5. TFF Tank WM-184 post-decontamination interior.

Figure A-6. TFF Tank WM-185 post-decontamination interior.

Figure A-7. TFF Tank WM-186 post-decontamination interior.

A-4. ALTERNATIVE RADIONUCLIDE CONCENTRATION CALCULATIONS

Radionuclide concentrations for piping were calculated using only the mass of the steel piping, but not the volume of the grout. These concentrations are shown in Tables A-13 and A-14. Radionuclide concentrations were also calculated for the grouted waste form in the tanks, without including the mass of the steel in the tanks, as shown in Tables A-15 through A-18.

Table A-13. Radionuclide concentrations in the piping (without grout) (Table 1 of 10 Code of Federal Regulations [CFR] 61.55).

Radionuclide[a]	Half-Life (yr)	Piping Inventory (Ci)[b]	Piping Inventory in Ci/m^3	Piping Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	5.3E–03		3.0E–01	100	0.0030
^{14}C	***5.7E+03***	***6.2E–08***	***2.8E–08***		***8***	***0.0000000035***
^{242}Cm	4.5E–01	1.7E–05		9.4E–04	20,000	0.000000047
^{129}I	***1.6E+07***	***9.7E–06***	***4.4E–06***		***0.08***	***0.000055***
^{94}Nb	***2.0E+04***	***2.6E–03***	***1.2E–03***		***0.2***	***0.0059***
^{59}Ni	***7.5E+04***	***3.1E–04***	***1.4E–04***		***220***	***0.00000065***
^{237}Np	2.1E+06	5.9E–04		3.4E–02	100	0.00034
^{238}Pu	8.8E+01	1.4E–01		8.2E+00	100	0.082
^{239}Pu	2.4E+04	4.3E–02		2.4E+00	100	0.024
^{240}Pu	7.0E+03	1.7E–02		9.6E–01	100	0.010
^{241}Pu	1.4E+01	2.4E–01		1.4E+01	3,500	0.0040
^{244}Pu	3.8E+05	1.2E–05		7.0E–04	100	0.0000070
^{99}Tc	***2.1E+05***	***9.6E–03***	***4.4E–03***		***3***	***0.0015***
Sum of the Fractions						**0.13**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining radionuclides are concentrations limits in units of nCi/g.
b. Radioactive decay to 2012.
c. Table 1 of 10 CFR 61.55.

Table A-14. Radionuclide concentrations in the piping (without grout) (Table 2 of 10 CFR 61.55).

Radionuclide	Half-Life (yr)	Piping Inventory (Ci)[a]	Piping Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.4E+01	6.5E+00	4,600	0.0014
^{63}Ni	1.0E+02	3.6E–02	1.6E–02	700	0.000023
^{90}Sr	2.9E+01	2.9E–01	1.3E–01	7,000	0.000019
Sum of the Fractions					**0.0015**

a. Radioactive decay to 2012.
b. Table 2 of 10 CFR 61.55.

Table A-15. Radionuclide concentrations in the final Tank WM-182 grouted waste form (calculated without the mass of the steel tank) (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	Tank Inventory (Ci)[b]	Tank Inventory in Ci/m^3	Tank Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	4.2E–01		8.9E–01	100	0.0089
^{14}C	***5.7E+03***	***5.0E–06***	***2.2E–08***		***8***	***0.0000000028***
^{242}Cm	4.5E–01	1.3E–03		2.8E–03	20,000	0.00000014
^{129}I	***1.6E+07***	***7.7E–04***	***3.4E–06***		***0.08***	***0.000043***
^{94}Nb	***2.0E+04***	***2.1E–01***	***9.1E–04***		***0.2***	***0.0046***
^{59}Ni	***7.5E+04***	***2.5E–02***	***1.1E–04***		***220***	***0.00000051***
^{237}Np	2.1E+06	4.7E–02		1.0E–01	100	0.0010
^{238}Pu	8.8E+01	1.1E+01		2.4E+01	100	0.24
^{239}Pu	2.4E+04	3.4E+00		7.2E+00	100	0.072
^{240}Pu	7.0E+03	1.4E+00		2.9E+00	100	0.029
^{241}Pu	1.4E+01	1.9E+01		4.1E+01	3,500	0.012
^{244}Pu	3.8E+05	9.9E–04		2.1E–03	100	0.000021
^{99}Tc	***2.1E+05***	***7.6E–01***	***3.4E–03***		***3***	***0.0011***
Sum of the Fractions						**0.37**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining radionuclides are concentration limits in units of nCi/g.
b. Radioactive decay to 2012.
c. Table 1 of 10 CFR 61.55.

Table A-16. Radionuclide concentrations in the final Tank WM-182 grouted waste form (calculated without the mass of the steel tank) (Table 2 of 10 CFR 61.55).

Radionuclide	Half-Life (yr)	Tank Inventory (Ci)[a]	Tank Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.1E+03	5.1E+00	4,600	0.0011
^{63}Ni	1.0E+02	2.9E+00	1.3E–02	700	0.000018
^{90}Sr	2.9E+01	2.3E+01	1.0E–01	7,000	0.000015
Sum of the Fractions					**0.0011**

a. Radioactive decay to 2012.
b. Table 2 of 10 CFR 61.55.

Table A-17. Radionuclide concentrations in a final 30,000-gal tank grouted waste form (calculated without the mass of the steel tank) (Table 1 of 10 CFR 61.55).

Radionuclide[a]	Half-Life (yr)	30,000-gal Tank Inventory (Ci)[b]	30,000-gal Tank Inventory in Ci/m^3	30,000-gal Tank Inventory in nCi/g	Class C Concentration Limit (Ci/m^3 or nCi/g)[c]	Fraction of Class C Concentration Limit
^{241}Am	4.3E+02	6.4E–03		5.3E–02	100	0.00053
^{14}C	***5.7E+03***	***1.1E–07***	***1.9E–09***		***8***	***0.00000000024***
^{242}Cm	4.5E–01	2.0E–05		1.7E–04	20,000	0.0000000083
^{129}I	***1.6E+07***	***1.2E–05***	***2.1E–07***		***0.08***	***0.0000026***
^{94}Nb	***2.0E+04***	***3.1E–03***	***5.5E–05***		***0.2***	***0.00027***
^{59}Ni	***7.5E+04***	***3.8E–04***	***6.7E–06***		***220***	***0.000000030***
^{237}Np	2.1E+06	7.1E–04		5.9E–03	100	0.000059
^{238}Pu	8.8E+01	1.7E–01		1.4E+00	100	0.014
^{239}Pu	2.4E+04	5.1E–02		4.3E–01	100	0.0043
^{240}Pu	7.0E+03	2.0E–02		1.7E–01	100	0.0017
^{241}Pu	1.4E+01	2.9E–01		2.5E+00	3,500	0.00070
^{244}Pu	3.8E+05	1.5E–05		1.2E–04	100	0.0000012
^{99}Tc	***2.1E+05***	***1.2E–02***	***2.0E–04***		***3***	***0.000068***
Sum of the Fractions						**0.022**

a. Radionuclides shown in ***bold italics*** are concentration limits in units of Ci/m^3; remaining radionuclides are concentration limits in units of nCi/g.
b. Radioactive decay to 2012.
c. Table 1 of 10 CFR 61.55.

Table A-18. Radionuclide concentrations in a final 30,000-gal tank grouted waste form (calculated without the mass of the steel tank) (Table 2 of 10 CFR 61.55).

Radionuclide	Half-Life (yr)	Tank Inventory (Ci)[a]	Tank Inventory in Ci/m^3	Class C Concentration Limit (Ci/m^3)[b]	Fraction of Class C Concentration Limit
^{137}Cs	3.0E+01	1.7E+01	3.0E–01	4,600	0.000066
^{63}Ni	1.0E+02	4.3E–02	7.6E–04	700	0.0000011
^{90}Sr	2.9E+01	4.5E–01	8.0E–03	7,000	0.0000011
Sum of the Fractions					**0.000068**

a. Radioactive decay to 2012.
b. Table 2 of 10 CFR 61.55.

A-5. REFERENCE

Wenzel, D. R., 2005, "Relative Inventories of Reactor-Produced Species in INTEC Waste Types," EDF-CRPD-001, Rev. 2, February 24, 2005.

Appendix B

Summary of Analysis for Selecting Tank Cleaning Technologies

Appendix B

Summary of Analysis for Selecting Tank Cleaning Technologies

In making the cleaning process technology selection for the Tank Farm Facility (TFF), the Idaho National Laboratory (INL) Site was part of a larger group called the Tanks Focus Area (TFA) technical team. The U.S. Department of Energy (DOE) commissioned this team to coordinate technology needs for the DOE complex-wide highly radioactive tank waste remediation problems. The team was established in 1995 for developing cleaning technologies and coordinating the tank cleanup effort among the DOE sites.

B-1. TECHNOLOGIES EVALUATED

The TFA coordinated with all DOE sites to develop cleaning processes based on site needs. The equipment developed had to fit inside the tanks, be compatible with the tank environment, and be able to clean the specific tank waste. Tank cleaning can be accomplished using either chemical or mechanical processes.

Chemical processes that the TFA developed include:

- Solids Washing—A chemical process for washing with Fenton's Reagent (a mixture of hydrogen peroxide with an Fe catalyst) that destroys ion exchange resin to release waste absorbed on the resin and allows it to be treated for disposal.

- Enhanced Solid Washing—A chemical process that involves a series of washes where tank waste is mixed with aqueous solutions containing sodium hydroxide, heated, cooled, and then the liquid, which contains the nonradioactive elements, is decanted.

- Chemical Cleaning—A process using various organic acids, possibly combined with caustic leaching to remove Al compounds, to dissolve portions of dense heel solids. By breaking up the solid mass, the resulting slurry can then be pumped out of the tank.

Mechanical processes include:

- Mixer Pumps—High-pressure pumps that intake and discharge solids in the tank bottoms to slurry the mixture and allow it to be pumped from the tank. Various systems were developed and tested, including:

  - A heavy waste retrieval system at the Oak Ridge Site

  - An advanced-design mixer pump at the Savannah River Site

  - A Flyght mixer at the Savannah River Site.

- Sluicing Systems—High-pressure water systems that slurry the solids and move it toward discharge pumps:

  - An enhanced sluicing system at the Hanford Site

  - A water mouse at the Savannah River Site

- A washball at the INL Site
- Directional nozzles at the INL Site
- A confined sluicing end effector at several DOE sites.

- Disposable Crawler—Commercially developed motorized treads that break up and mobilize the solids. A sluicer mounted on top of the motorized treads then uses a high-pressure water jet to move the loosened material toward the transfer pump.

- Mechanical Arms—Robotic arms installed through tank risers that are capable of deploying in-tank surveillance, confined sluicing, inspection, and waste analysis tools called end effectors:
  - Light-duty utility arm at the INL Site, the Hanford Site, and the Oak Ridge Site
  - Advanced Waste Retrieval System at the West Valley Demonstration Project.

B-2. INL SITE TECHNOLOGY EVALUATION

Because TFF tanks were considered with all tanks in the entire DOE complex, much of the work accomplished to clean other site tanks was applicable to the TFF tanks. Lessons were learned from technologies developed for other sites, even when the technology was not directly applicable to the TFF tanks. As a result of the TFA and the complex-wide effort at tank cleaning, the INL Site did not need to accomplish a lot of basic research on tank cleaning technologies. Rather, the INL Site was able to build on proven technologies and improve them for INL Site-specific needs. However, basic differences exist between the TFF waste and waste from other sites. The INL Site waste is acidic and contains few solids, whereas waste from other sites is non-acidic and contains many solids, most of which precipitated when the waste was neutralized for storage. Because of these differences, not all TFA-developed technologies were applicable at the INL Site. The INL Site participated with the TFA, evaluated TFA and other technologies, and determined that mechanical, rather than chemical, cleaning was applicable.

Through evaluation and the process of elimination, the INL Site determined that chemical processes were not practical for cleaning the TFF tanks. Caustic recycle, solids washing, and saltcake dissolution were developed for the neutralized waste at other DOE sites and did not apply to the TFF acid waste. Chemical cleaning was not practical for TFF waste because an acid that was strong enough to dissolve the solids could cause tank corrosion. Washing the solids in a basic solution could cause more precipitates, which would further aggravate the solids problem. Based on this evaluation, the INL Site did not pursue chemical cleaning.

The INL Site determined that some, but not all, mechanical cleaning systems were applicable for cleaning tanks. Mixer pumps were developed to slurry large solids volumes and were not appropriate for INL Site use because the TFF tanks contain relatively few solids. Mechanical robots that clean material from the tank floor were not practical because they could not crawl over the cooling coils on some of the TFF tanks. However, sluicing systems and mechanical arms were compatible with the TFF tanks. Sluicing systems were practical for removing the small quantity of solids from the tank walls and for slurrying the relatively thin layer of solids on the tank bottoms. A mechanical arm could hold a video camera for in-tank inspections and could operate equipment for sampling tank residuals.

The washball and a directional nozzle were determined to be best for cleaning the tank walls and slurrying solids on the tank bottoms. The washball was developed commercially by the chemical and oil industries for tank cleaning. The TFA introduced it to the DOE complex, and the INL Site has deployed it

for cleaning the TFF tanks. The INL Site borrowed directional nozzle technology from the TFA and other sites to develop a remotely operable sluicing system that consists of a high-pressure spray nozzle, lights, and video camera that can be extended into a tank on a long pipe (PNNL 2001). The INL Site has deployed two high-pressure spray-nozzle sluicing systems for tank cleaning.

The INL Site tank cleaning system was tested in a full-scale mockup tank using simulated waste prior to deploying the unit for use in the tanks. The washball/directional nozzle tank cleaning system and the modified steam-jet pumping system were used to transfer slurried solid and liquid waste from the tanks. (The steam jet was modified by cutting the steam-supply line and installing a new steam jet lower in the tanks.) As a result of this development work and in-tank use of the washball, directional nozzle, light-duty utility arm, and modified steam-jet pumping system, the INL Site has created an efficient and effective tank cleaning system.

B-3. INL SITE TANK CLEANING

Tanks that will be closed by the TFF Closure Project include 11 300,000-gal tanks and four 30,000-gal tanks. One of the 300,000-gal tanks has never been used to store high-level waste and contains only a small quantity of slightly contaminated water. Contaminated lines, encasements, and valve boxes will also be cleaned, and for the purposes of calculating radionuclide removal costs, their cleaning will be considered as part of tank cleaning. During washball and directional nozzle operations, the steam-jet ejectors will operate to remove the waste-containing slurry from the tank. The goal of tank cleaning is to remove as much waste as practical. As an indication of cleaning effectiveness during this operation, radiation levels will be monitored on the steam-jet transport line (see Attachment B-1). Based on experience, when the radiation readings reach a certain level, the cleaning operation will be stopped and the tanks will be sampled to ascertain that they meet requirements for tank closure. However, if radiation levels are still dropping significantly, cleaning will continue. When radiation levels decrease to the lowest constant value, cleaning will be stopped and the tanks inspected and sampled to verify closure requirements are met (Kimmitt 2002).

B-4. WORKER DOSE

The INL Site has conducted tank cleaning on Tank WM-182 at the TFF. Cleaning operations for Tank WM-182 have included removing old equipment, installing tank cleaning and sampling equipment, and flushing with approximately 100,000 gal of water. Based on radiation exposure information on tank cleaning operations (Jacobson 2002) and a review of radiation work permit electronic dosimetry results for January 1, 2002, through June 15, 2005 (Martin 2005), the following is the estimated radiation exposure for cleaning a TFF tank.

Tank WM-182 cleaning occurred from mid-June to October 1, 2002. Filling Tank WM-182 with grout will result in a small additional exposure. The radiation exposures recorded in the Radiation Control Information Management System for 2002 tank closure activities for Tank WM-182 cleaning are a total of 611 mrem, including 15 mrem for removing equipment and preparing Tank WM-183 for a washball test (Jacobson 2002). Radiation exposures recorded for all TFF work performed January 1, 2002, through June 15, 2005, further reveal a total of 4,931 person-mrem. Maintenance-related process line and valve work for this period account for 2,568 person-mrem or about 52% of the total worker exposure. If the entire balance of 2,363 person-mrem is attributed to tank cleaning for the seven tanks, this yields an average of 338 person-mrem for each tank. Since 338 person-mrem per tank is conservatively derived from all tank activities averaged over the 3.5 years and seven tanks, it is concluded that using the actual exposure of 611 person-mrem from cleaning the most contaminated tank (WM-182) is a reasonable dose projection for future tank cleaning (Martin 2005).

The program's as low as reasonably achievable goal for 2002 was 1.93 person-rem, of which the TFF Closure Project contributed 0.611 person-rem. Twenty-three personnel involved directly with tank cleaning received a radiation dose from TFF closure activities. The maximum exposure any worker has received from TFF closure activities is 117 mrem.

Based on this information, the following is concluded:

- The average radiation exposure that will be experienced for cleaning and closing each TFF tank is expected to total about 650 mrem for all occupational exposure
- The exposure per person for cleaning a TFF tank will be about 650 mrem divided by 23 people, which is about 30 mrem per person
- Maximum radiation exposure for an individual worker is estimated to be 120 mrem for cleaning a single TFF tank.

Worker dose for tank cleaning is minimal because all cleaning is accomplished remotely. Worker exposures would be limited to equipment installation and operation and maintenance activities on contaminated equipment. Worker exposure per tank is estimated to total approximately 650 mrem for 23 workers (30 mrem per worker), which results in a total exposure of about 7.15 rem for cleaning 11 300,000-gal tanks. Worker doses for complete tank removal are significant because the action would require a large project to excavate the tanks, cut them up, and package them for disposal. Worker exposure for complete removal of the TFF tank system is estimated to be 1,070 mrem/yr/worker for an average of 326 workers/year for an estimated 26 years for a total exposure of over 9,000 rem (INEEL 1998).

B-5. REFERENCES

INEEL, 1998, *ICPP Tank Farm Closure Study,* INEEL/EXT-97-01204, Rev. 0, Appendix A, February 1998.

Jacobson, V. L., INEEL, to K. Quigley, INEEL, October 30, 2002, "Radiation Exposures from Tank WM-182 Cleaning."

Kimmitt, R. R., INEEL, 2002, to V. L. Jacobson, INEEL, October 30, 2002, "Tank WM-182 Cleaning Effectiveness."

Martin, J. R., Portage, Inc., to K. Quigley, INL, July 7, 2005, "Review of Radiation Exposures for INEEL INTEC Tank Farm – January 2002–June 2005."

PNNL, 2001, *Technical Review of Retrieval and Closure Plans for the INEEL INTEC Tank Farm Facility*, PNNL UC-721, Pacific Northwest National Laboratory, Richland, Washington, September 2001.

Attachment B-1

Tank Cleanout Effectiveness

The following tank cleaning information is from Tank WM-182 cleaning operations. Since Tank WM-182 was the first INTEC TFF tank that was cleaned to levels required before grouting, these data are presented as representative of the tank cleaning process.

Tank WM-182 Cleaning Results

Throughout summer 2002, Tank WM-182 was cleaned as part of a State of Idaho-approved closure plan. Cleaning operations took place on 18 days over a 3- to 4-month period. The primary goal of the cleaning was to remove radioactivity and chemical contaminants from the tank. Now that the radioactivity has been removed to the maximum extent practical, the remaining contents will be grouted in place within the tank. Mockup tests have shown that the grouting process also moves much of the remaining liquids and solids to the ejector pump so they can be removed. (For a complete explanation of data and results, see Kimmitt [2002].)

Background

By July 1999, Tank WM-182 had been emptied to heel levels by removing as much sodium-bearing waste as possible. Based on the quantities of solids and liquid estimated from the in-tank video inspection and on sample results, the total Tank WM-182 radionuclide content was estimated to be 14,000 Ci in the solids and 13,800 Ci in the liquid before cleaning operations began.

Cleaning was accomplished by modifying the in-tank steam jet to lower it to within 3 cm (1 in.) of the tank floor, washing with clean high-pressure water, and pumping the slurried solids and diluted liquid from the tank. Tank WM-182 was cleaned using a washball and directional spray nozzles to rinse and slurry the solids. During cleaning operations, tank closure personnel operated the modified steam jet to remove as much of the slurried solids as practical.

Data Collection

Radioactivity being pumped from Tank WM-182 was monitored in the discharge piping in Valve Box C2. The detector was a calibrated, unshielded Geiger-Mueller counter mounted near the pipe. Output from the counter was recorded at counts per minute at periodic time intervals. No measurement or estimate of detector efficiency was available. The count rate data were provided to project personnel in the form of charts. Figure Att-B-1-1 is an example of the discharge pipe readings.

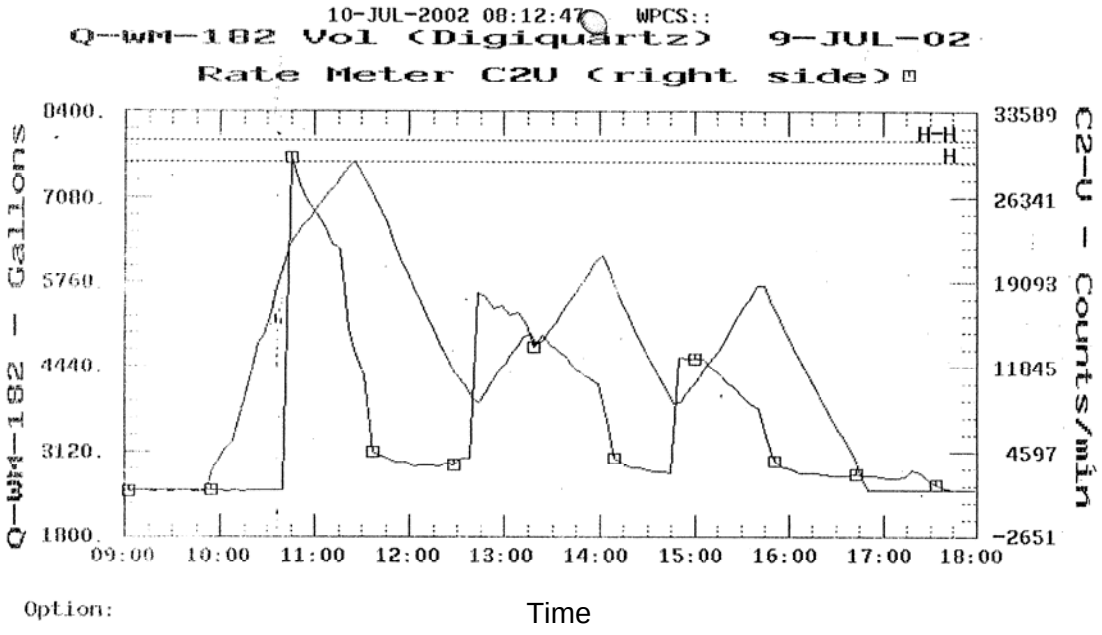

Figure Att-B-1-1. Discharge pipe radiation readings.

No discharge volume flow rate data were available. Therefore, the ejector was assumed to have operated continuously at 50 gpm during each cleaning session. Flow to the washball and directional spray nozzles was interrupted periodically to prevent the tank fluid level from rising above the desired range.

Data Analysis

To determine how the radioactivity content of Tank WM-182 was decreasing, an estimate of the radioactivity concentration of material being pumped from the vessel was necessary. The count rate was typically noted at 15-min intervals. From these values and the flow rate, a concentration term was calculated and plotted against cumulative volume of liquid pumped from the vessel. A background level was required to be subtracted from the count rate information. The background level was the starting count rate on each day before cleaning operations began. Figure Att-B-1-2 shows the resulting curve.

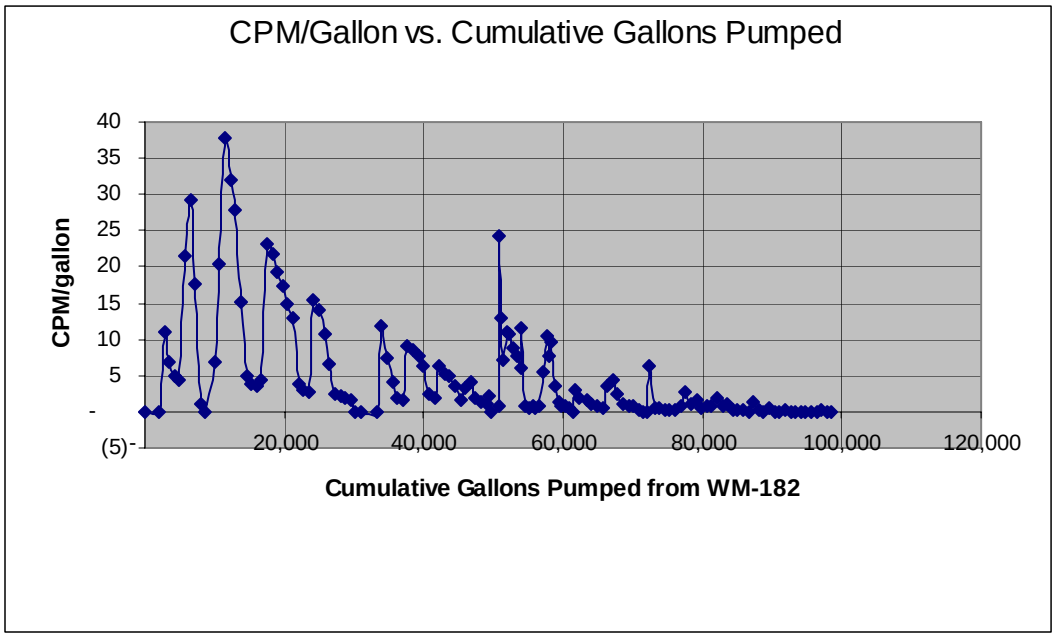

Figure Att-B-1-2. Radioactivity concentration [counts per minute per gallon] curve.

This figure shows that initially, a great deal of variability in the radioactivity per unit volume of water pumped existed. Drops occurred when the washwater was shut off and the suspended solids settled to the bottom of the tank. Once the washwater flow was restarted, the solids were re-suspended and pumped from the vessel. At first, only the washball was used, and it effectively stirred the solids. However, after approximately 30,000 gal had been introduced into the tank during several cleaning cycles, this device became far less effective. At approximately 50,000-gal cumulative volume introduced, two directional nozzles were substituted for the washball. Solids removal efficiency increased immediately but eventually tapered off again. By the time 90,000 gal of water had been pumped, essentially no additional radioactivity was being removed with the washwater. The gross Geiger-Mueller monitoring provided the cleaning personnel and management with a method to evaluate the need for additional cleaning.

Appendix C

Tank Farm Facility Grout Design Mix and Quality Assurance/Quality Control Testing

Appendix C

Tank Farm Facility Grout Design Mix and Quality Assurance/Quality Control Testing

The closure process for each tank includes removing most of the bulk sodium-bearing waste, grouting the tanks, vaults, and associated piping. (A final cover to be placed over the entire area upon completion of closure activities is anticipated, but this cover is not considered in analyses for this document.) The period between the first tank closure and final closure of the area is expected to be a minimum of 20 years. To meet closure requirements, each tank will be grouted with a Portland cement and fly ash-based grout. Final grout formulas are still being considered. The grout formulas may change, but the general performance objectives will remain the same. Testing of the grout in the simulated conditions expected at tank closure has been performed and is documented in the *Grout Analysis Summary Report* (SAIC 2000).

C-1. PERFORMANCE OBJECTIVES

Using grout in the tank closures provides long-term structural stability of the waste under the expected disposal conditions. If the final waste form is structurally stable, it will meet the following criteria (per 10 Code of Federal Regulations [CFR] 61):

- Prevent excessive subsidence, settlement, or deformation
- Minimize water infiltration
- Prevent radionuclide release from waste form disintegration
- Minimize the likelihood of waste intrusion.

Using grout in the tank closure process also provides a mechanical means for repositioning tank heels such that the waste residuals can more easily be removed from the tanks. The grout then serves to integrate and encapsulate the waste residuals. To accomplish this action, the grout must have the appropriate consistency and density to displace the waste. The grout must remain sufficiently flowable during placement.

The above uses are, in various instances, mutually exclusive. Specifically, increasing flowability is generally accompanied by a strength reduction and increased susceptibility to cracking. Based on the 1999 Idaho Nuclear Technology and Engineering Center grouting mockups, it is understood that flowability versus strength issues will be balanced by engineering the grout mix designs and placement sequencing (INEEL 2000, Book 1, Volume I, Chapter 16).

C-2. GROUT MIX DESIGN

Two basic grout mixtures are being considered for Tank Farm Facility (TFF) closure. The first (pipe grout) is a mixture of Portland cement, fly ash, water, and water-reducing admixtures. This mixture will be used to fill piping, small vessels, or other equipment that may require a very fluid grout. The other type of grout is called either tank grout or vault grout. They have the same basic mixture but differing volumes of constituents to account for differences in the required strength and the slump or flowability of the grout. The grout is expected to exhibit strongly reducing conditions, as do most concrete systems (DOE-ID 2003). Current TFF analysis concludes that reducing conditions in the grout are not necessary to

demonstrate compliance with performance objectives (Portage 2005). As such, the addition of reducing agents such as fly ash or other materials to ensure reducing conditions is being evaluated. The vault grout is required to flow around at least half the circumference of the existing 15.2-m (50-ft) diameter tanks.

The flowability or fluidity of the mixtures is the main requirement for the grouts. This requirement has been measured in the past using standard concrete slump tests (ASTM International [ASTM] C 143/ C 143M). Approximately 23 cm (9 in.) of slump is required for all of the mixtures, and the pipe grout requires slumps of approximately 28 cm (11 in.). The tank grout must be able to move the tank waste residuals (both liquid and some solids) to locations where the wastes can more easily be removed from the tanks. For this reason, the tank grout needs to be less fluid than the other grout mixtures. Adjusting the quantity of water present in the mixtures appears to be the easiest way to obtain the desired fluidity.

The base mix design for the tank and vault engineering grout pours is as follows (EDF-1464, 2000):

Cement	320 lb	Type I & II cement
Pozzolan Class F	640 lb	Fly ash
Fine aggregate	2,200 lb	Sand
Water	Up to 433 lb	(52 gal)
Medium-range water reducer	Up to 64 oz	
High-range water reducer	Up to 96 oz	

The base mix design for the controlled low-strength material, which will be used to fill the tanks and vaults above the engineering grout pour volumes, is as follows:

Cement	75–100 lb	Type I & II cement
Pozzolan Class F	200–300 lb	Fly ash
Fine aggregate	2,750–3,100 lb	Sand
Water	300–400 lb	(36–48 gal)
Medium-range water reducer	Up to 64 oz	
High-range water reducer	Up to 96 oz	

Grout placement tests associated with the first stage of the conceptual design shows that grout with a slump of more than 28 cm (11 in.) will flow halfway around the circumference of a 15.2-m (50-ft) diameter tank. Testing results indicate that all the mixtures are flowable and will achieve 28-day compressive strengths of at least 2,000 psi. Because of the high ash content of the mixtures, they continue to gain significant strength for a longer time period than is normal for typical construction grouts. The 56-day compressive strengths are approximately 1,000 psi stronger than the 28-day compressive strengths.

The base mix design for the pipe grout is as follows (EDF-1464, 2000):

Cement	680 lb	Type I & II cement
Pozzolan Class F	1,600 lb	Fly ash
Water	Up to 800 lb	(96 gal)
Medium-range water reducer	Up to 32 oz	
High-range water reducer	Up to 64 oz	

The 28- and 56-day compressive strengths for the pipe grout mixture (using later tests based on 91 gal of water per yd^3) are 3,360 and 4,680 psi, respectively. The lowest 28-day test compressive strength for the pipe grout mixture tests is 2,400 psi (EDF-1464, 2000).

The slump of the pipe grout as tested by ASTM C 143/C 143M was in excess of 29 cm (11.4 in.). For larger pipes, it may be possible to reduce the water content of the mixture, and thus, reduce the shrinkage and bleed water associated with the high water content. Bleed water amounts from past tests of the mixture were low.

C-3. MANUFACTURING QUALITY CONTROL

The grout supplier will provide and manufacture grout according to the following requirements (INEEL 2000, Section 16):

- Cement, Sand, and Fly Ash
  - ASTM C 150 Standard Specification for Portland Cement
  - ASTM C 618 Standard Specification for Coal Fly Ash or Calcined Natural Pozzolan for Use as a Mineral Admixture in Portland Cement Concrete
- Chemical Admixtures
  - ASTM C 494/C 494M Standard Specification for Chemical Admixtures for Concrete
  - ASTM C 1017/C 1017M Standard Specification for Chemical Admixtures for Use in Producing Flowing Concrete.

C-4. MANUFACTURING QUALITY ASSURANCE

Manufacturing quality assurance (QA) will be conducted by an engineer and generally includes the following:

- Review documentation stating the supplier's qualifications, capabilities, licenses, certifications, etc.
- Periodic inspections of the manufacturing plant
- Verification that substantive requirements of manufacturing quality control are being met

- Construction quality control.

The construction quality control procedures are normally established by the contractor and include activities such as delivery cycles, truck waiting time limitations, mixing revolutions, water and admixture measurements, and any other procedures that are used to meet the project specifications consistently (INEEL 2000, Section 16).

C-5. CONSTRUCTION QUALITY ASSURANCE

Construction QA activities conducted by the engineer at the project site will consist of the following:

- Observe and record the contractor's activities and construction quality control procedures
- Receive batch tickets, and randomly verify that delivery meets project specifications
- Provide adequate grout samples for field and laboratory testing per applicable ASTM standards
- Field test grout per applicable ASTM standards.

The following ASTM standards or equivalents for controlled low-strength material shall be used for construction QA:

- Sampling Procedures
  - ASTM C 31/C 31M Standard Practice for Making and Curing Concrete Test Specimens in the Field
  - ASTM C 172 Standard Practice for Sampling Freshly Mixed Concrete
- Grout Testing
  - ASTM C 143/C 143M Standard Test Method for Slump of Hydraulic-Cement Concrete
  - ASTM C 939 Standard Test Method for Flow of Grout for Preplaced-Aggregate Concrete (Flow Cone Method)
  - ASTM C 1064/C 1064M Standard Test Method for Temperature of Freshly Mixed Hydraulic-Cement Concrete.

Periodic tests will be conducted at the engineer's discretion to verify that the delivered grout used throughout the TFF closure project has compressive strength properties similar to preliminary design values and remains consistent over time. Tests on representative samples will be conducted per ASTM C 39, *Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens* (INEEL 2000, Section 16).

C-6. QUALITY ASSURANCE TESTING SUMMARY

Table C-1 is a summary of the QA tests and procedures that are projected for the TFF closure grouting project. They are based on concrete construction industry standards of practice.

Table C-1. Summary of QA Tests for Grouting (INEEL 2000, Section 16).

QA Test or Procedure	Standards	Frequency and Time, Other Notes
Review supplier's qualifications	NA	Once, prior to beginning work
Inspect batch plant	NA	Quarterly for the project duration, more frequently if grout inconsistencies are noted
Check compliance with manufacturing standards	ASTM C 150 ASTM C 494/ C 494M ASTM C 618 ASTM C 1017/ C 1017M	These checks will be performed as part of the batch plant inspections
Observe and record contractor's activities and quality control procedures	NA	Continuously through project duration, prepare daily logbooks, weekly reports, and monthly reports
Measure grout temperature	ASTM C 1064/ C 1064M	One per day
Measure grout slump or time of efflux	ASTM C 143/ C 143M ASTM C 939	One per day
Collect grout test specimens	ASTM C 31/C 31M ASTM C 172	Collect representative cylinders for further testing if engineer determines necessary.
Perform compressive strength tests	ASTM C 39/C 39M	Collect representative cylinders for further testing if engineer determines necessary.

NA = Not applicable.

The water-to-cement ratio is the chief factor for determining grout strength. Therefore, careful measuring and testing of the mix during manufacturing and upon arrival at the job site is important to the final product. Because real-time physical characterization of the grout is needed prior to placement, grout samples should be collected from the beginning of the truck discharge prior to placement. These samples should be tested for slump (ASTM C 143/C 143M) and/or time of efflux (ASTM C 939). Although it is standard procedure to measure air content (ASTM C 231) in conjunction with field sampling of concrete, in this case air content measurements may be waived. Prior test data indicate very low values of air content for the proposed mix design, and these values are not expected to increase during the TFF closure. In addition, entrained air is used to give grout freeze and thaw resistance, and frost is not anticipated in this design application.

C-7. REFERENCES

10 CFR 61, 2004, "Licensing Requirements for Land Disposal of Radioactive Waste," *Code of Federal Regulations*, Office of the Federal Register, January 1, 2004.

ASTM C 31/C 31M, 2003, *Standard Practice for Making and Curing Concrete Test Specimens in the Field*, Rev. A, ASTM International, February 10, 2003.

ASTM C 39/C 39M, 2004, *Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens*, Rev. A, ASTM International, November 1, 2004.

ASTM C 143/C 143M, 2003, *Standard Test Method for Slump of Hydraulic-Cement Concrete*, ASTM International, July 10, 2003.

ASTM C 150, 2004, *Standard Specification for Portland Cement*, Rev. A, ASTM International, July 1, 2004.

ASTM C 172, 2004, *Standard Practice for Sampling Freshly Mixed Concrete*, ASTM International, June 1, 2004.

ASTM C 231, 2004, *Standard Test Method for Air Content of Freshly Mixed Concrete by the Pressure Method,* ASTM International, July 1, 2004.

ASTM C 494/C 494M, 2004, *Standard Specification for Chemical Admixtures for Concrete*, ASTM International, January 1, 2004.

ASTM C 618, 2003, *Standard Specification for Coal Fly Ash or Calcined Natural Pozzolan for Use in Portland Cement Concrete*, ASTM International, January 10, 2003.

ASTM C 939, 2002, *Standard Test Method for Flow of Grout for Preplaced-Aggregate Concrete (Flow Cone Method)*, ASTM International, December 10, 2002.

ASTM C 1017/C 1017M, 2003, *Standard Specification for Chemical Admixtures for Use in Producing Flowing Concrete*, ASTM International, December 1, 2003.

ASTM C 1064/C 1064M, 2005, *Standard Test Method for Temperature of Freshly Mixed Hydraulic-Cement Concrete*, ASTM International, January 1, 2005.

DOE-ID, 2003, *Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory*, DOE/ID-10966, Rev. 1, April 2003 (Errata December 2, 2003).

EDF-1464, 2000, "INTEC Tank Farm Closure Grout Mix Design," April 14, 2000.

INEEL, 2000, *INTEC Tank Farm Facility Closure Conceptual Design Report*, Project File No. 015722, September 29, 2000.

Portage, 2005a, "Evaluation of ^{99}Tc Drinking Water Dose for Oxidizing Sorption Coefficient in the Tank Grout," PEI-EDF-1024, Rev. 0, Portage, Inc., Idaho Falls, Idaho, August 2005.

SAIC, 2000, *Grout Analysis Summary Report,* Science Applications International Corporation, August 11, 2000.

Appendix D

Management Control Systems

Appendix D

Management Control Systems

This appendix summarizes the management control systems applicable to the Tank Farm Facility (TFF) Section 3116 Determination process. They are designed to ensure that both primary project objectives are met and an optimum margin of safety for protection of personnel, the public, and the environment are achieved. The management controls implemented by the field managers for these elements ensure that the Section 3116 determination criteria are met for disposal of the residual waste and TFF tank system at the Idaho National Laboratory (INL) Site. The following elements are addressed:

- Procedures
- Quality assurance (QA)
- Document and record control
- Training and qualifications.

D-1. PROCEDURES

The Idaho Nuclear Technology and Engineering Center (INTEC) TFF draft 3116 Determination was developed using formal processes and methods. Existing INL Site policies, programs, and procedures were used to manage and implement many of the INL Site activities that support the Section 3116 Determination process. Implementing documents and procedures were also used for tank inventory sampling, data collection, analyses, and other activities performed in support of the Section 3116 Determination process. This subsection discusses the key documents used for the management and performance of the Section 3116 Determination activities.

The INL Site Document Management Control System presents written instructions for preparing, reviewing, approving, maintaining, and distributing documents and changes to documents. The Document Management Control System applies to controlled documents. The system was developed to prescribe processes, specify requirements, and establish design as it relates to TFF closure activities, including development of the TFF draft 3116 Determination.

The *Idaho National Engineering and Environmental Laboratory Waste Acceptance Criteria* (DOE-ID 2005) defines U.S. Department of Energy Idaho Operations Office (DOE Idaho) requirements for characterizing, packaging, and documenting reusable property, recyclable materials, and waste to be received by the INL Site. The scope of the waste acceptance criteria (WAC) includes requirements applicable to the following radioactive waste classifications: low-level waste, transuranic waste, and high-level waste. The WAC also specifies requirements for identifying and managing hazardous and nonhazardous wastes under the Hazardous Waste Management Act (State of Idaho 1983)/Resource Conservation and Recovery Act (42 United States Code [USC] 6901 et seq., 1976). The WAC requires that each generator of radioactive waste provides assurance that appropriate sections of the acceptance criteria and applicable requirements are met.

Management control procedures (MCPs) are controlled implementing documents that prescribe administrative processes to be performed to support TFF closure and development of the draft 3116 Determination. Specific implementing documents that pertain to TFF closure and the TFF draft 3116 Determination are listed in QA program requirements documents listed in the *Quality and Requirements*

Management Program Documents (INL 2005). The INL Site implementing procedures and documents prescribe how work is to be performed.

D-2. QUALITY ASSURANCE

The U.S. Department of Energy (DOE) and its contractors shall develop and maintain a QA program for radioactive waste management facilities, operations, and activities that meet the requirements of 10 Code of Federal Regulations (CFR) 830, Subpart A, "Quality Assurance Requirements," and DOE Order 414.1B, "Quality Assurance" (2004), as applicable.

The TFF draft 3116 Determination was developed under a QA program that ensures the validity of the information used to make the determination. This subsection describes the QA programs applicable to the TFF draft 3116 Determination to ensure compliance with the *Quality and Requirements Management Program Documents* (INL 2005). Figure D-1 illustrates the relationship of the various QA programs and documents discussed below.

The DOE Idaho establishes QA requirements for the INL Site management and operating (M&O) contractor through the INL M&O contract with CH2M-WG Idaho, LLC (CWI). The INL Site M&O *Quality and Requirements Management Program Documents* describe CWI's QA program and are based on DOE Order 414.1B, "Quality Assurance" (2004), and 10 CFR 830, Subpart A, "Quality Assurance Requirements." The *Quality and Requirements Management Program Documents* apply to M&O organizations responsible for achieving, maintaining, and verifying the quality of items and activities in support of facilities, programs, and projects; and to companies performing work for CWI, as specified in procurement contracts.

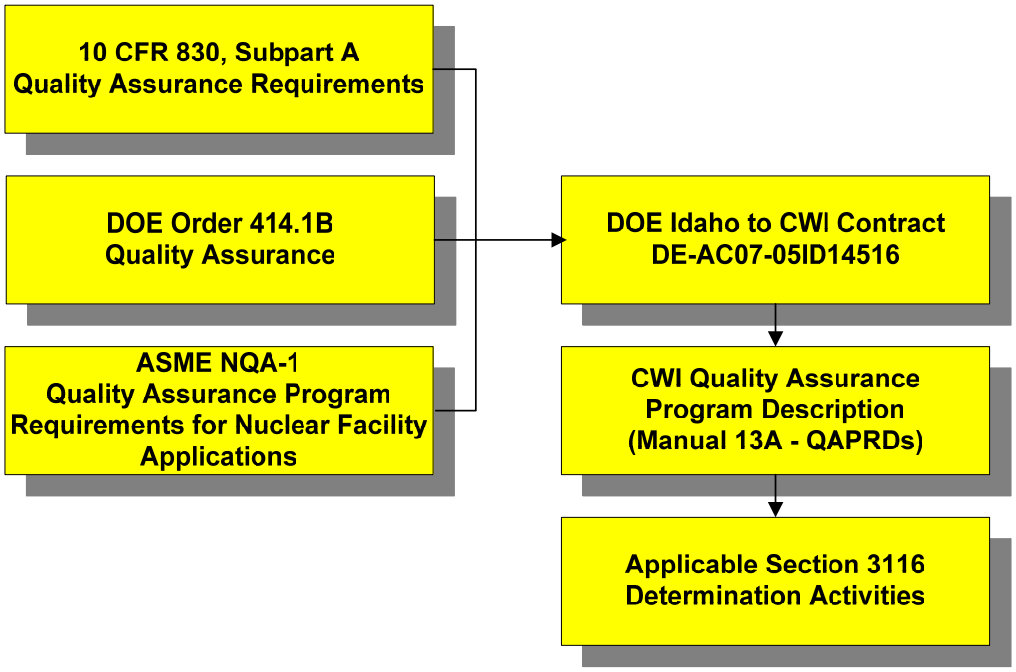

Figure D-1. Quality assurance program hierarchy.

Table D-1 identifies the specific American Society of Mechanical Engineers (ASME) NQA-1 (ASME 1997) elements that CWI has determined are applicable to the waste process. The waste process

QA requirements will be evaluated against the WAC to ensure that QA program requirements are met for waste storage, transportation, and disposal.

Details for implementing the 10 NQA-1 elements listed in Table D-1 can be found in the "Quality Program Plan for the High-Level Waste Program Office" (PLN-627, 2002). This plan describes the QA program for high-level waste activities managed by the INL Site, including TFF closure and Section 3116 Determination activities. The CWI implements its QA program using a graded approach. The graded approach incorporates safety categories that identify the relative importance of an item or activity to the consequence of failure, should failure of the item or activity occur. The MCPs are used to facilitate implementation of the graded approach and the assignment of quality levels to structures, systems, and components and activities.

D-3. DOCUMENT AND RECORD CONTROL

Records management systems ensure that records important to safety and quality are generated, reviewed, approved, collected, and maintained. The management system provides controls so that records accurately reflect completed work and facility conditions and comply with applicable statutory or contractual requirements.

The INL Site MCPs incorporate the requirements of DOE Order 200.1, "Information Management Program" (1996), and DOE Order 414.1B, "Quality Assurance" (2004). Schedules for records retention and disposition are in accordance with the General Records Schedule of the National Archives and Records Administration and other approved records schedules. The MCPs include instructions for retention, protection, preservation, changes, traceability, accountability, and retrievability of records. They also provide controls to ensure records are legible, accurate, complete, retrievable, and validated by authorized personnel.

Records are stored and maintained to minimize the risk of damage, larceny, vandalism, or deterioration. Active records are not sent to records holding facilities but are stored in a facility where the records may be readily accessed.

Table D-1. ASME NQA-1 applicability (ASME 1997).

ASME NQA-1 1997 Element Number	Description
1	Organization
2	Quality Assurance Program
3	Design Control
4	Procurement Document Control
5	Instructions, Procedures, & Drawings
6	Document Control
7	Control of Purchase Items and Services
16	Corrective Action
17	Quality Assurance Records
18	Audits

D-4. TRAINING AND QUALIFICATIONS

The INL Site training program focuses on providing employees with the knowledge and skills necessary to perform tasks that meet acceptance criteria. The INL Site training, with the assistance of subject-matter experts, is responsible for analyzing, designing, developing, implementing, and evaluating training programs and processes. The MCPs detail the instructional processes used, including self-study, computer, and video-based training, instructor-led training, and on-the-job training.

The DOE Idaho and CWI define training and qualification requirements for selected positions or job categories by considering the level of knowledge and skills required to perform tasks. Training plans are developed to guide development of skills and knowledge necessary for employees to meet requirements of specific job categories. Training and qualification requirements are established and periodically reviewed to ensure that requirements continue to reflect training needs.

The INL Site MCPs incorporate the requirements of DOE Order 5480.20A, "Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities" (1994). The MCPs describe personnel selection requirements and training, qualification, certification, and continued training processes. Procedures specify the frequency for which training is needed. The DOE Idaho and CWI determine and document when personnel are suitably qualified to accomplish assigned tasks.

Personnel that perform draft 3116 Determination development, review, approval, and revision functions receive training in the applicable scope, purpose, and objectives of the Section 3116 Determination process and the specific QA objectives of the assigned task before performing Section 3116 Determination process activities. Personnel also receive training on applicable implementing procedures used in the performance of the task.

This training includes appropriate subject material from the following documents:

- DOE Order 435.1, "Radioactive Waste Management" (2001)
- DOE Manual 435.1-1, "Radioactive Waste Management Manual" (2001)
- DOE Guidance 435.1-1, "Implementation Guide for use with DOE M 435.1-1" (1999)
- 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste," Subpart C, "Performance Objectives," and Subpart D, "Technical Requirements for Land Disposal Facilities," 61.55, "Waste Classification"
- *Idaho National Engineering and Environmental Laboratory Waste Acceptance Criteria* (DOE-ID 2005)
- QA program requirements documents listed in the *Quality and Requirements Management Program Documents* (INL 2005).

Training is documented and training records are maintained in accordance with INL Site MCPs. Qualifications of CWI personnel supporting the Section 3116 Determination development process are documented, and appropriate records are maintained. Subcontractors supporting tank closure and Section 3116 Determination activities are selected based on company and personnel qualifications and experience. All training records are available for review.

D-5. QUALITY ASSURANCE FOR DATA COLLECTION

Quality assurance can be defined as an integrated system of management activities involving planning, implementation, documentation, assessment, reporting, and quality improvement to ensure that a process, item, or service is the type and quality needed and expected by the customer (EPA 2000a). This quality system, according to the U.S. Environmental Protection Agency (EPA 2000a), can be viewed as having three tiered levels of hierarchy: policy, organization or program, and project. Policy, the highest level, is comprised of the mandated regulatory drivers that must be addressed. The middle level, organization or program, addresses the management and implementation component of the individual quality system. And finally, at the bottom level, are the project-specific components that are applied to ensure that the needs of the organization are met. The TFF closure project used the data quality objective (DQO) and data quality assessment (DQA) process during all phases of data collection and assessment.

At the project level, three stages define the project's life cycle: planning, implementation, and assessment. Systematic planning, the first key component to QA, is based on a common sense, graded approach to ensure that the level of detail in planning is commensurate with the importance and intended use of the work and the available resources. An effective approach to plan data collection efforts, to establish acceptance criteria for the quality of data collected, and to develop an appropriate data collection design to support decision-making is the DQO process.

D-5.1 The Data Quality Objective Process

The DQO process helps to focus studies, to clarify vague objectives, and to limit the number of decisions that will be made. The DQO process breaks the planning approach to develop sampling designs for data collection activities that support decision-making into seven steps.

1. State the Problem: Define the problem, identify the planning team, and examine budget and schedule.

2. Identify the Decision: State decision, identify study question, and define alternative actions.

3. Identify the Inputs to the Decision: Identify information needed for the decision (information sources, basis for action level, sampling and analysis method).

4. Define the Boundaries of the Study: Specify sample characteristic, define spatial or temporal limits, and establish units of decision-making.

5. Develop a Decision Rule: Define statistical parameter (e.g., mean, median), specify action level, and develop logic for action.

6. Specify Tolerable Limits on Decision Error: Set acceptable limits for decision errors relative to the consequences (e.g., health effects, costs).

7. Optimize the Design for Obtaining Data: Select resource-effective sampling and analysis plan that meets the performance criteria.

The DQO process is intended to be both flexible and iterative, allowing the planning team (senior program staff, technical experts, managers, data users, regulators, and stakeholders) to incorporate new information and to incorporate outputs from previous steps into subsequent planning processes. The DQO process is best suited to problems that require making a decision between two clear alternatives. The final

outcome is a design for collecting data (such as the number of samples to collect, when, where, and how to collect samples) and the limits on the probabilities of making decision errors.

The DQOs are defined as both qualitative and quantitative statements that clarify study objectives, define the appropriate type(s) of data to be collected, and specify tolerable levels of potential decision errors. They form the basis for establishing the quality and quantity of data needed to support decisions and the performance criteria that limit the probabilities of decision errors.

The first five steps in the DQO process are primarily focused on identifying quality criteria (e.g., the nature of the problem that has initiated the study) a conceptual model of the environmental hazard to be investigated, the decisions that need to be made and the order of priority for resolving them, the type of data needed (i.e., geographic area, environmental medium, overall timing of data collection, etc.), and a decision rule (a theoretical "*If…then…*" statement) defining how the data will be used to choose among alternative actions.

The sixth step defines quantitative criteria, expressed as limits on the probability or risk of making a decision error that the decision-maker can tolerate. A decision error occurs when the sample data set misleads the decision-maker into the wrong decision and wrong response action. For example, declaring a site contaminated and requiring further remediation when, in fact, the clean-up activities have been successful.

The seventh step is used to develop a data collection strategy based on the criteria developed in the first six steps. The outputs of the DQO process are used to perform a DQA.

D-5.2 Data Quality Assessment

Data quality assessment is a statistical and scientific evaluation of the data set used to determine the adequacy of the data set for its intended use. In general, the DQA provides a scientific and statistical evaluation of data to determine if acquired data are of the right type, quality, and quantity to support their intended use. The DQA process is designed around the key idea that data quality, as a concept, is only meaningful when it directly relates to the intended use of the data (EPA 2000b). Two primary questions can be answered using the DQA process:

1. Does the quality of the data permit decisions to be made with the desired degree of confidence?

2. How well can the sampling design be expected to perform over a wide range of possible outcomes? That is, can the sampling design strategy be expected to perform well in a similar study with the same degree of confidence even if the actual measurements are different than those obtained in the present study?

The first question addresses the immediate needs of the study. If the assessment shows the data are of sufficient quality, it allows for sound decision-making (i.e., unambiguous data allows for decisions with the desired level of confidence as specified during data acquistion planning). Conversely, if the data do not provide sufficiently strong evidence to support one decision over another, then appropriate data analysis can alert the decision-maker to the degree of ambiguity in the data. If this is the case, an informed decision can be made about how to proceed. For example, based on the data obtained, more data may be collected or the decision-maker may make a decision knowing there is a greater-than-desired uncertainty in the decision.

The second question addressed by the DQA process is possible future applications of the study. After the DQA is completed, a determination is made as to how well the sampling design may perform at

a different location given the different environmental conditions and outcomes that likely occur at the other location. Because environmental conditions vary from location to location, it is important to examine the sampling design over a large range of possible settings to ensure it will be adequate for other scenarios.

Data quality assessment consists of both data validation and analysis of the validated data set. Data validation is employed to ensure all sampling and analysis protocols were followed properly. Data analysis is performed to the level of confidence decision-makers can place in the data set. The DQA process is comprised of the following five steps:

1. Review the DQOs and sampling design

2. Conduct a preliminary data review

3. Select a statistical test

4. Verify the assumptions of the selected test

5. Draw conclusions from the data.

The DQA process is intended to be an iterative process that promotes an understanding of how well the data satisfy their intended use. It is completed by progressing in a logical and efficient manner. It is important to note that because data collection provides an estimate of the value of interest (a decision-maker can never know the true value of the item of interest), the DQA cannot absolutely prove that the DQOs set forth in the planning phase have been achieved. Rather, the DQA evaluates the risk of making a wrong decision about the item of interest. Ultimately, the DQA process provides the decision-maker with a quantitative means of evaluating these risks and a means to progress to the next phase of the program.

D-6. REFERENCES

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